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2	1	Cog.Eng. B. W. Hall for B. W. Hall 8/7/95			
2	1	Cog. Mgr. J. G. Kristofzski J. G. Kristofzski 8/7/95			
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This document was reviewed following the procedures described in WHC-CM-3-4 and is:

APPROVED FOR PUBLIC RELEASE

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Signature 8/7/95

Name: T. T. Tran, LATA

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Signature

Organization/Charge Code 75310/N4162

7. Abstract

This document summarizes the information on the historicall uses, present status, and the sampling and analysis results of waste stored in tank 241-BX-105. This report supports the requirements of Tri-Party Agreement Milestone M-44-08.

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Revision 0

Tank Characterization Report for Single-Shell Tank 241-BX-105

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Office of Environmental Restoration and
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Westinghouse Hanford Company
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EXECUTIVE SUMMARY

This tank characterization report summarizes the information on the historical sources, present status, and the sampling and analysis results of the waste stored in the single-shell underground storage tank 241-BX-105. This report supports the requirement of the *Hanford Federal Facility Agreement and Consent Order*, Milestone M-44-08 (Ecology et al. 1994).

Tank 241-BX-105 is one of 12 single-shell tanks located in the BX Tank Farm in the Hanford Site's 200 East Area. The tank is the second in the tank 241-BX-104, -105, -106 cascade series. The tank went into service in 1949 when it received metal waste produced in the bismuth phosphate process. This waste was cascaded from tank 241-BX-104. Other waste types received by the tank during its service life were uranium recovery waste, Plutonium-Uranium Extraction Plant (PUREX) cladding waste, ion-exchange waste, evaporator bottoms, B Plant low-level waste, double-shell slurry feed, and concentrated reduction and oxidation (REDOX) wastes. For several years the tank operated as an evaporator feed storage tank and later became an active receiver of salt well wastes pumped from other tanks in the BX Tank Farm or from tanks in the BY Tank Farm.

Tank 241-BX-105 was deactivated in 1980. The tank was designated interim stabilized after most of the supernatant was removed in March of 1981. Additional supernatant pumping and intrusion prevention were completed in 1986. The tank is currently classified

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as sound. The tank is not on any watch list and has no unreviewed safety questions associated with it.

The description and status of the tank are summarized in the Table ES-1 and Figure ES-1. The tank, which has an operating capacity of 2,010 kL (530 kgal), presently contains approximately 193 kL of waste, consisting mostly of sludge, with some salt cake and supernatant. The temperature of the tank has been stable, around 21 °C, since the majority of the supernatant was pumped from the tank in 1986.

This report summarizes two sampling and analysis events. The first sampling event involved the taking of two core samples from the tank, one each from risers 1 and 8, which are located on opposite sides of the tank near the perimeter. The core samples were tested for a wide range of chemical and radiochemical analytes. The analytical results from this first sampling event have been used to estimate tank waste compositions and to project total analyte inventories for the tank. The inventory estimates are provided to support retrieval efforts. In the second sampling event, auger samples were taken from risers 2 and 6, located adjacent to risers 1 and 8, respectively. The analytical results from the auger samples have been used to evaluate the tank's contents against various safety screening criteria defined in *Tank Safety Screening Data Quality Objective*, WHC-SD-WM-SP-004 (Babad and Redus 1994). Specifically, the auger samples were tested for exothermic reactions using differential scanning calorimetry, for water content using thermogravimetric analysis, and for fissile radioisotope content using total alpha analysis.

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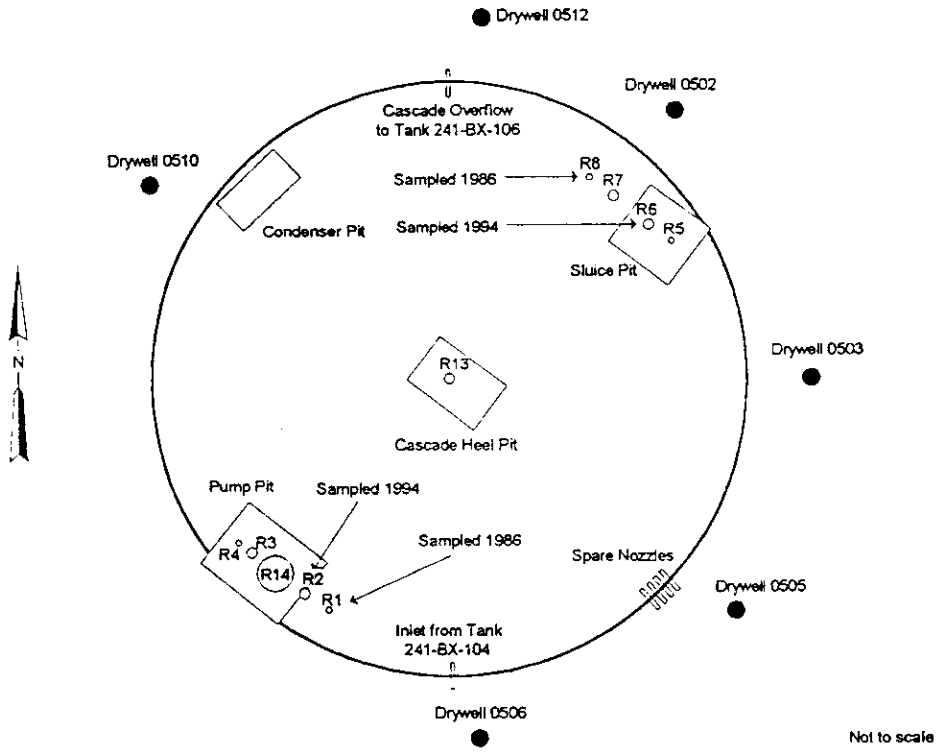
Table ES-1. Summary of Tank 241-BX-105.

Tank description	
Type:	Single-shell
Constructed:	1947
In-service:	1949
Diameter:	22.86 m (75 ft)
Usable depth:	4.88 m (16 ft)
Operating capacity:	2,000 kL (530 kgal)
Bottom shape:	Dish
Ventilation:	Passive
Tank status	
Watch list:	Non-watch list
Contents:	Non-complexed waste
Estimated total waste volume:	193 kL (51 kgal)
Estimated salt cake volume:	11.4 kL (3 kgal)
Estimated sludge volume:	163 kL (43 kgal)
Estimated supernatant volume:	18.9 kL (5 kgal)
Estimated interstitial liquid volume:	22.7 kL (6 kgal)
FIC surface level (CASS):	63 cm (24.8 in.)
Temperature:	Stabilized around 21 °C (70 °F)
Integrity:	Sound
Interim stabilized:	1981
Intrusion prevention:	1986
Service status:	Not in service

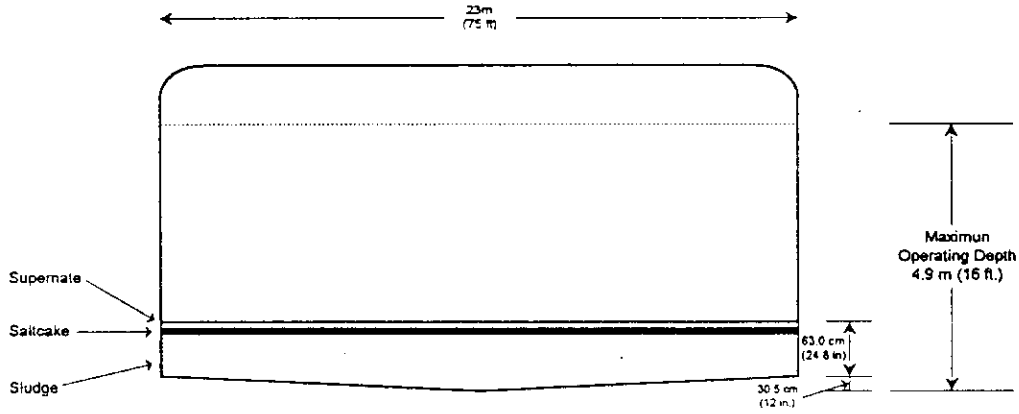
FIC = Food Instrument Corporation.
CASS = Computer Automated Surveillance System

Figure ES-1. Tank 241-BX-105.

Tank 241-BX-105



Waste Profile of Tank 241-BX-105



Total Tank Volume: 2,010 kL (530 kgal)
Waste Volume (Oct. 1994): 193 kL (51 kgal)
Saltcake Volume (Oct. 1994): 11.4 kL (3 kgal)
Sludge Volume (Oct. 1994): 163 kL (43 kgal)
Supernate Volume (Oct. 1994): 18.9 kL (5 kgal)
Interstitial Liquid Volume (Oct. 1994): 22.7 kL (6 kgal)

Not to scale

Data from the 1986 core sampling event are considered in this report to be valid to estimate the current contents of the tank because the core samples were taken after the tank was taken out of service and interim stabilized. The 1986 core samples were separated into liquid and solid phases before analysis. Table ES-2 summarizes the results of the analyses on the centrifuged solids (sludge) and centrifuged liquid phases and provides inventory estimates for both phases. The concentration values in the table represent the average of the results on the riser 1 and riser 8 composites. The sludge and liquid inventory estimates were determined based on the analyte concentration estimates, the total estimated volume of the tank's contents, and the measured liquids and solids volumes and weights following centrifugation. The uncertainty of the inventory values is potentially large because of the small quantity of sample tested and because of poor sample recovery for one of the cores.

The inventory estimates in Table ES-2 are consistent with the tank's operating history. A comparision is made in this report between the analytically based inventory estimates and hostorically based inventory estimates. The high concentrations of uranium, sodium, and PO_4 measured in the tank waste are consistent with the receipt of metal waste from B Plant in the late 1940s and early 1950s and with the receipt of uranium recovery waste in 1956. The high aluminum and sodium concentrations are consistent with the receipt of PUREX cladding waste from 1963 through 1968 and with the receipt of REDOX waste in 1980. The high ^{137}Cs content is likely due to the receipt of IX waste between 1973 and 1976.

Table ES-2. Single-Shell Tank 241-BX-105, Concentrations and Inventories For Analytes of Concern.

Physical properties	Sludge result		Liquid results	
Density	1.69 g/mL		1.29 g/mL	
Percent water[a]	57%		68.7%	
Heat load	327 W (1120 Btu/h)			
Chemical constituents	Sludge concentration	Sludge inventory	Liquid concentration	Liquid inventory
Metals	(μg/g)	(kg)	(mg/L)	(kg)
Aluminum	33,900	7,660	5930	349
Bismuth	3,200	723	0.783	4.6 E-02
Calcium	6,210	1,400	144	8.48
Chromium	8,620	1,950	249	14.7
Iron	5,940	1,340	4.37	0.257
Sodium	110,000	24,900	124,000	7,300
Silicon	41,500	9,380	139	8.19
Uranium	4,410	997	4.51 E-03	2.66 E-04
Organics	(μg C/g)	(kg)	(g/L)	(kg)
Total organic carbon	2,780	628	9.44	556
Anions	(μg/g)	(kg)	(mg/L)	(kg)
NO_3^-	33,300	7,530	1.90	0.112
PO_4^{-b}	61,000	13,800	5,210	307
Radionuclides	(μCi/g)	(Ci)	(μCi/L)	(Ci)
^{241}Am	0.993	224	24.5	1.44
^{14}C	1.29 E-03	0.292	0.608	3.58 E-02
^{60}Co	0.158	35.7	177	10.4
^{137}Cs	54.1	12,200	1.75 E+05	10,300
^{129}I	3.00 E-05	6.78 E-03	5.10 E-02	3.00 E-03
$^{239/240}Pu$	0.279	63.1	17.1	1.01
^{90}Sr	136	30,700	11,200	660
^{99}Tc	3.35 E-02	7.57	114	6.71

1 Ci = 3.7 E+10 Bq.

[a]Based on heating of sample to 400 °C, may be biased high.

[b]Conversion from inductively coupled plasma spectroscopy data.

The core samples were homogenized prior to analysis, so the vertical heterogeneity of the waste could not be assessed. Based on the tank's operating history, however, vertical heterogeneity in the tank waste is almost assured. There was considerable variability between the riser 1 and riser 8 core sample results, indicating that the waste may be laterally heterogeneous as well.

None of the differential scanning calorimetry (DSC) results on the 1994 auger subsamples exhibited exotherms. Using both the centrifuged sludge and drainable liquid analyses from the 1986 core sampling event, the total organic carbon (TOC) content for the tank waste is estimated to be 0.98% dry weight. This value is well below the 5% TOC (dry weight) criterion established by the organic safety program (Babad, Blacker, and Redus 1994). The 1994 DSC results and the 1986 TOC results indicate, therefore, that the waste does not contain excessive amounts of fuel.

The average wt% water of all the thermogravimetric analyses (TGA) performed on the 1994 auger samples was 12.6%. Large amounts of moisture reduce the potential for propagating exothermic reactions in the waste. The minimum water content specified in the Tank Safety Screening Data Quality Objective is 17 wt%. However, since no exotherms or significant fuel sources were detected in the waste, there is no indication that a potential for runaway reactions exists in the tank.

The highest total alpha result from the 1994 auger subsamples was 0.589 μCi/g. The highest plutonium concentration measured in the 1986 solids composites was 0.474 μCi/g. These values are well below the established criterion of 1 g/L specified in the Tank Safety Screening data quality objective (DQO) (by a factor of 70 or more). Assuming all the plutonium is the ^{239}Pu isotope, and using an average measured solids density of 1.69 g/mL, the 1 g/L limit converts to 36.4 μCi/g. The total alpha values exceed the 100 nCi/g TRU designation limit and thus the tank should be considered as containing TRU waste.

Based on the 1986 core sample data, the heat generation rate for the tank is estimated to be 327 W (1,120 Btu/h). This is far below the 40,000 Btu/h criterion for distinguishing a high heat load tank from a low heat load tank (Hanlon 1995).

Based on the information summarized above, the waste in tank 241-BX-105 does not appear to pose any immediate safety concerns. Although the waste exceeds the TRU limit, no immediate safety concern are created. The TRU limit is an operational segregation rule which plays a larger role during transfer and mixing of waste streams. It is recommended, if future samples are taken, that additional analyses be considered to provide better estimates of lateral and vertical heterogeneity and reduce the uncertainty in fuel and water content measurements.

REFERENCES

Babad, H., and K. S. Redus, 1994, *Tank Safety Screening Data Quality Objective*, WHC-SD-WM-SP-004, Westinghouse Hanford Company, Richland, Washington.

Babad, H., S. M. Blacker, and K. S. Redus, 1994, *Data Quality Objective to Support Resolution of the Organic Fuel Rich Tank Safety Issue*, WHC-SD-WM-DQO-006, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

Ecology, EPA, and DOE, 1994, *Hanford Federal Facility Agreement and Consent Order, fourth amendment*, Washington State Department of Ecology, U.S. Environmental Protection Agency, and U.S. Department of Energy, Olympia, Washington.

Hanlon, B.L., 1995, *Waste Tank Summary for Month Ending January 31, 1995*, WHC-EP-0182-82, Westinghouse Hanford Company, Richland, Washington.

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CASS	Computer Automated Surveillance System
DQO	data quality objective
DSC	differential scanning calorimetry
FIC	Food Instrument Corporation
HEDTA	hydroxyethylethylenediaminetriacetic acid
ICP	inductively coupled plasma (spectroscopy)
ND	not detected
NPH	normal paraffin hydrocarbon
PUREX	Plutonium-Uranium Extraction (Plant)
REDOX	reduction and oxidation
RPD	relative percent difference
TGA	thermogravimetric analysis
TOC	total organic carbon
TRU	Transuranic (waste)
WHC	Westinghouse Hanford Company

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1.0 INTRODUCTION

1.1 PURPOSE

The purpose of this report is to provide an overview of Single-Shell Tank 241-BX-105 and its waste contents. This report briefly describes the design of Tank BX-105, summarizes the tank's waste transfer history, provides an historical estimate of the tank's waste contents, summarizes surveillance information on the tank, and summarizes and evaluates the results of two sampling events that occurred after the tank was interim stabilized in 1980. The first sampling event discussed in this report occurred in 1986, when two core samples were removed from risers on opposite sides of the tank. The results from this sampling event are used in this report to project best-estimate analyte inventories for the entire waste volume of the tank. The analyte inventories are provided to support retrieval, pretreatment, and vitrification development activities. The analyte inventories projected from the 1986 core sample results are compared and evaluated against historical analyte inventories developed from best estimate waste transfer information.

The second sampling event discussed in this report pertains to the removal of two 51 cm (20-in.) auger samples from the tank in 1994. The analytical results from the auger samples are used to evaluate the tank against criteria defined in the *Tank Safety Screening Data Quality Objective* (Babad and Redus 1994). Pertinent results from the 1986 event are compared to the 1994 results and used to augment conclusions drawn on safety screening issues.

This report supports the requirements of the *Hanford Federal Facility Agreement and Consent Order*, Milestone M-44-08 (Ecology et al. 1994).

1.2 SCOPE

As discussed above, this report is intended to summarize information concerning the design, status, operating history, and sampling of Tank BX-105. Discussion of sampling activities are limited to those that occurred after the tank was interim stabilized in 1980. The 1986 core sampling event discussed in this report includes extensive chemical and radiochemical characterization information. The 1986 characterization results are generally lacking in Quality Assurance information. Under constraints of the Hanford Federal Facility Agreement and Consent Order (Ecology, et al. 1994), pre-1989 characterization data may not be considered valid for some applications. However, as the 1986 core sampling event provides the only radiochemical and chemical data available, it is used in this report to estimate tank waste inventories. The 1994 auger samples were taken to satisfy safety

screening data quality objectives (Babad and Redus 1994). The 1994 data therefore only include thermogravimetric analysis (TGA), differential scanning calorimetry (DSC), and total alpha results. The 1986 data are used, where pertinent, to augment conclusions drawn from the 1994 results concerning safety screening issues. The 1986 data is not used alone to evaluate the safety screening objectives.

Tank BX-105 is not on the flammable gas watch list. This report includes no information on vapor space sampling and analysis to determine the composition of the tank head space gases. When vaper space sampling results are available, they will be incorporated into a future revision to this document.

2.0 HISTORICAL TANK INFORMATION

Tank 241-BX-105 was removed from service in 1980 and interim stabilized in 1981. Supernatant was pumped from the tank in August of 1986 and intrusion prevention measures were incorporated in September of 1986. The most current tank volume status is provided. Tank history includes tank design information, waste transfer history, and waste temperature and level surveillance information.

2.1 TANK STATUS

Recent FIC gauge level readings taken from riser 1 of the tank indicate a waste depth of 63 cm (24.8 in). Assuming a level surface across the diameter of the tank, the level reading of 63 cm corresponds to a waste volume of 306 kL (this does not correspond to the volume of 193 kL recorded in Hanlon [1995], see Section 2.4.1). Core samples recovered from risers 1 and 8 seem to confirm the waste depth and volume indicated by the FIC gauge. The discrepancy between the waste volume estimate and the FIC level gauge readings is not explained in Hanlon (1995). A relatively large volume of supernatant (61 kL) was pumped from the center of the tank in 1986 without having much effect on the surface level at the perimeter of the tank below riser 1. This would seem to indicate that the solids layer in the tank may have assumed a dish shape, with the level in the center lower than the level at the edges, due to pumping from the center.

Tank BX-105 is classified as sound and is considered to be a non-watch list tank. The tank has been inactive since 1980 and was interim stabilized in March 1981. Intrusion prevention was completed in September 1986, and there are no unreviewed safety questions associated with the tank.

2.2 TANK DESIGN

Tank 241-BX-105 is a single-shell, underground storage tank consisting of a carbon steel tank within a reinforced concrete shell and dome (Brevick et al. 1994a). It has a capacity of 2,010 kL (530 kgal), a diameter of 22.9 m (75 ft), and a depth of about 4.9 m (16 ft) (Brevick et al. 1994). The basic design of tank 241-BX-105 is presented in Figure 2-1. The tank has a dished bottom. The tank is one of 12 located in the BX Tank Farm, which was constructed between 1946 and 1947 in the Hanford Site's 200 East Area (see the *Tank Characterization Reference Guide* [De Lorenzo et al. 1994] for information about the BX Tank Farm). Figure 2-2 depicts the location of the BX Tank Farm.

Figure 2-1. Location of the BX Tank Farm.

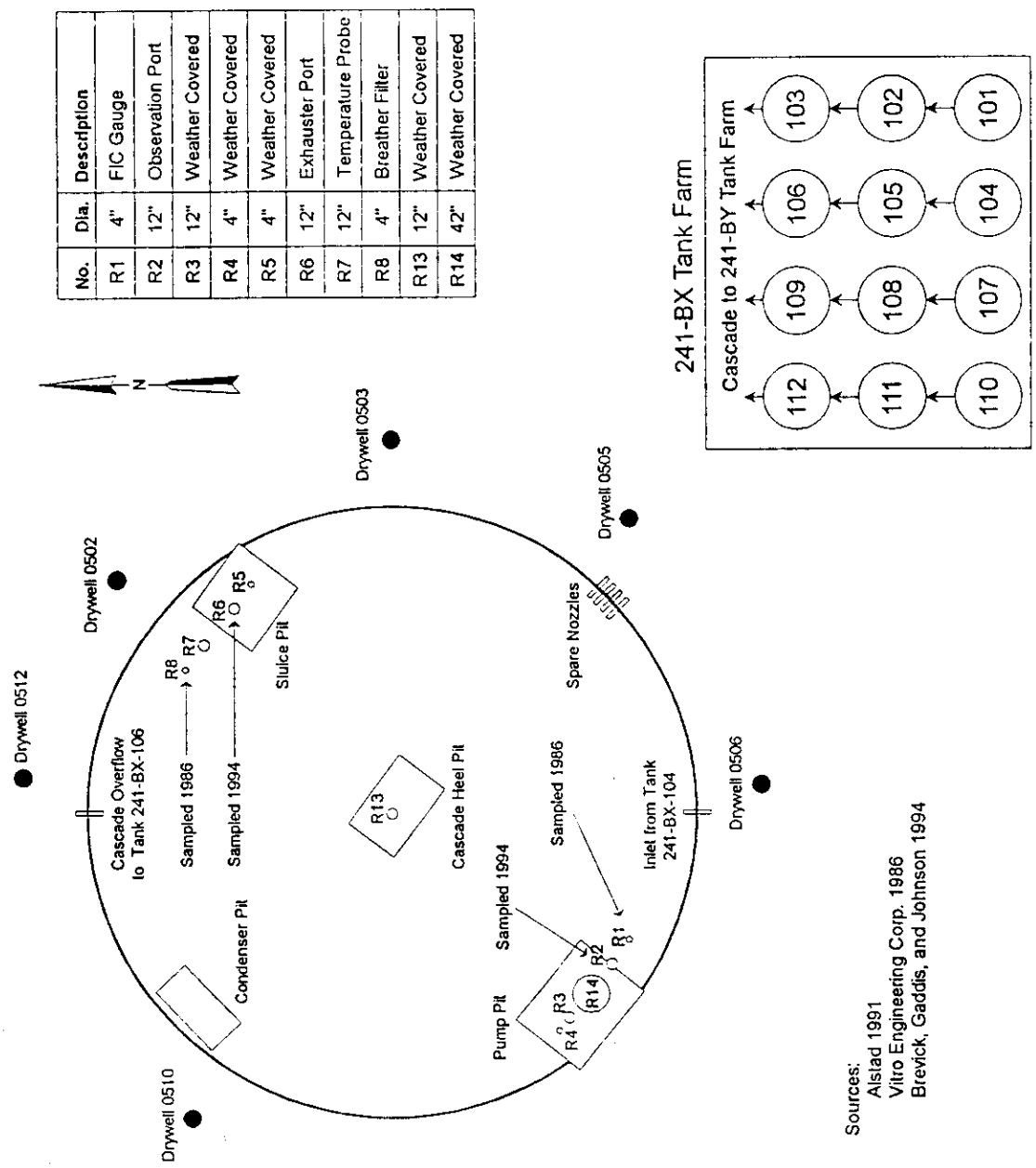
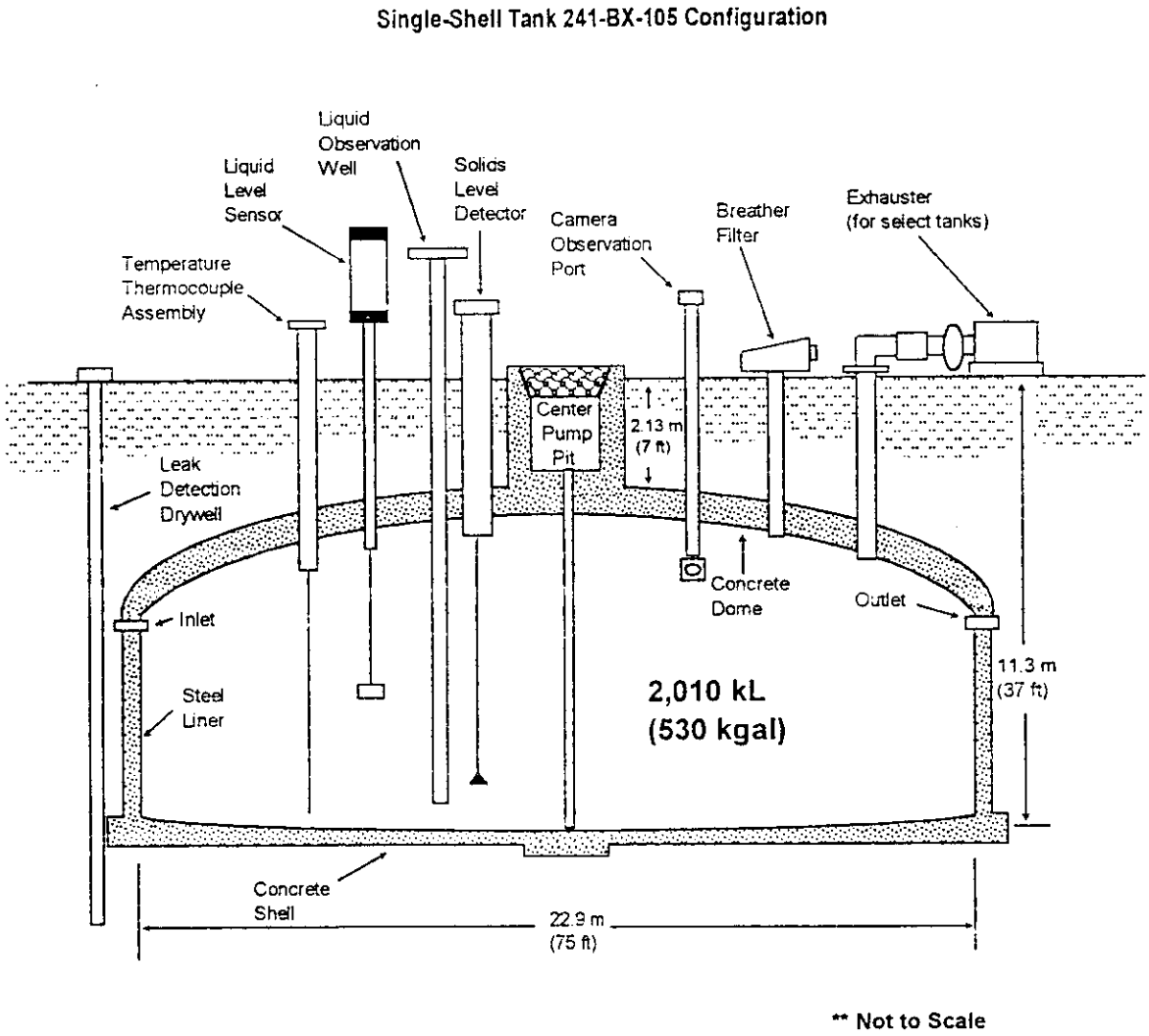


Figure 2-2. Basic Design of Tank 241-BX-105.



The tanks in the BX Tank Farm are similar in design to the first Hanford Site tanks (farms B, C, T, and U) constructed in 1943 through 1944. The tanks are arranged in four cascade groups. A cascade group consists of three tanks in step configuration. Tank 241-BX-105 is the second tank in a cascade group consisting of tanks 241-BX-104, -105, and -106. The cascade overflow height is approximately 478 cm (188 in.) from the tank bottom.

Instruments access the tank through risers and monitor the temperature, liquid level, and other bulk tank characteristics. The positions of these risers are shown in Figure 2-3. The tank is equipped with an automated liquid indicator device (through riser 1) to measure surface level readings. The tank is passively ventilated (Brevick et al. 1994).

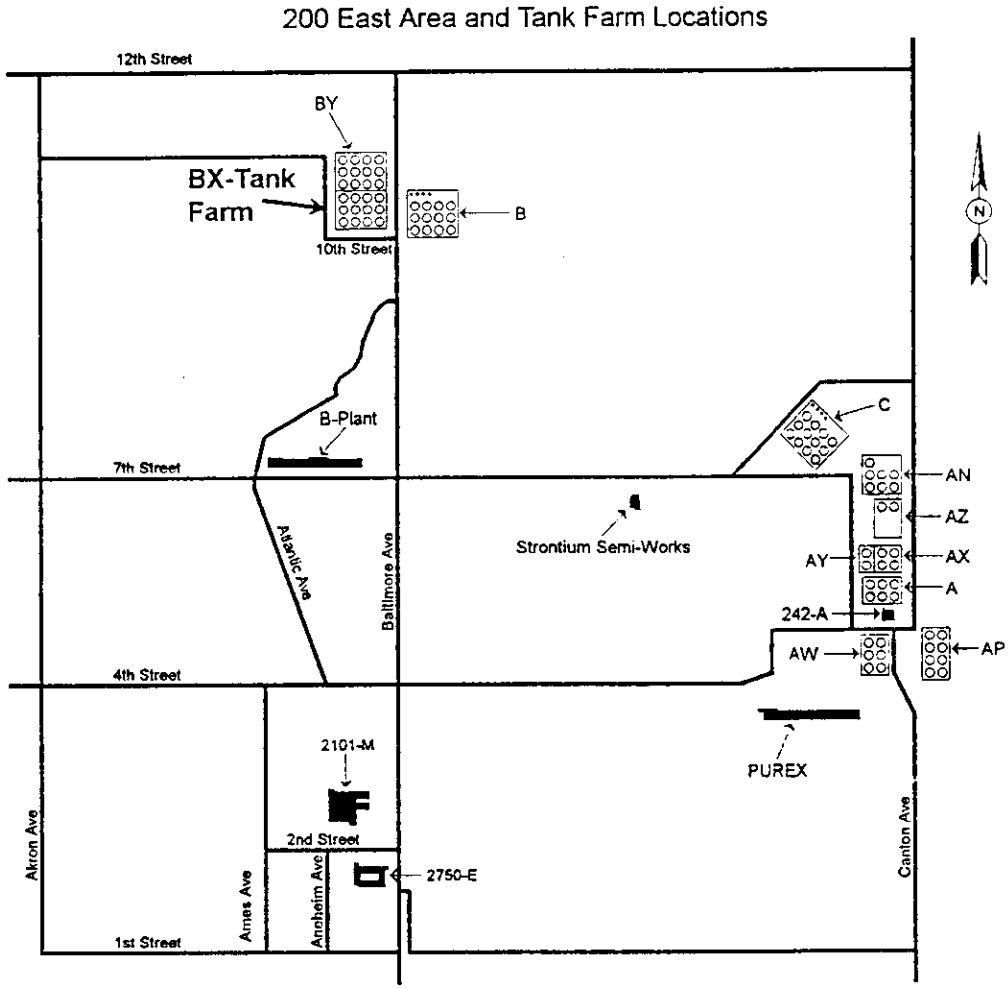
2.3 PROCESS KNOWLEDGE

The BX Tank Farm cascade group received B Plant metal waste. Tank 241-BX-105 was filled by the cascade line with metal waste in 1949. The tank was sluiced for uranium recovery in 1954 and was used as a leach tank to recover uranium from 1955 received to 1956. It received U Plant waste from 1956 to 1962. From 1963 to 1964 the tank received cladding waste from the PUREX Plant. The tank received ion-exchange waste from 1969 to 1974. Between 1974 and 1980, the tank contained evaporator bottoms waste, ion-exchange waste, and evaporator feed. The tank became inactive in late 1980. Supernatant was pumped from tank 241-BX-105 and interim stabilization was completed in March 1981. Additional supernatant was pumped from the tank and intrusion prevention was completed in September 1986.

2.3.1 Waste Transfer History

Tank 241-BX-105 was the second tank in a cascade that received metal waste from the bismuth phosphate process employed during the late 1940's and early 1950's at B Plant (unless noted otherwise, information in this section comes from *Waste Status and Transaction Record Summary for the Northeast Quadrant*, WHC-SD-WM-T1-615 [Agnew 1994a]). Metal waste resulted from the initial plutonium precipitation step. It contained high amounts of uranium, sodium, and phosphate (Agnew 1994b). The first tank in this cascade, tank 241-BX-104, began filling in January 1949. Waste began overflowing into tank 241-BX-105 through a cascade tie in April of the same year. Tank 241-BX-105 was filled in September 1949 and waste began cascading to tank 241-BX-106. Although most of the metal waste solids were expected to settle in the first tank of the cascade (tank 241-BX-104), some of the insoluble material still entered tank 241-BX-105 and settled there (Rodenhizer 1987). About 337 kL (89 kgal) of metal waste sludge were measured in the tank in 1954.

Figure 2-3. Riser Configuration for Tank 241-BX-105.



The tanks in this cascade were sluiced for uranium recovery during 1954-55 because of the high uranium content in the metal waste. Sluicing operations in the BX Tank Farm centered around tank 241-BX-105 in January, February, and May 1955 (Rodenhizer 1987). Sluicing, operations were completed by August of the same year, leaving a metal waste remnant heel in the tank of 49.2 kL (13 kgal). Liquid waste was then added to the tank to leach uranium from metal waste remnants (Anderson 1990; Agnew 1994a).

The tank was nearly refilled in 1956 with uranium recovery waste that had been stored in the BY Tank Farm. Much of this supernatant was pumped to either a trench or to tank 241-BY-102 the following year. Uranium recovery waste was comparatively high in concentrations of uranium, sodium, phosphate, and sulfate (Agnew 1994b). During this time the tank also received significant amounts of waste from sources that are now unknown.

From 1963 to 1968 the tank was filled with PUREX cladding waste supernatant that had been stored previously in tank 241-C-109, and supernatant from ion-exchange waste, cladding waste, and evaporator bottoms from tank 241-BX-104. PUREX cladding waste was produced during the dissolution of aluminum fuel cladding (Agnew 1994b). It was comparatively high in aluminum, sodium, and hydroxide. Ion-exchange waste resulted from cesium recovery operations at B Plant. It contained significant concentrations of cesium, uranium, and citrate, and a trace of trisodium hydroxyethylethylenediaminetriacetic acid (HEDTA). Evaporator bottoms is a salt slurry high in nitrates and sodium.

Tank 241-BX-105 received small additions of waste water from 1972 to 1976. In 1973, the tank received B Plant low-level waste and ion-exchange waste that had been stored in tanks 241-BX-104 and 241-BX-108. B Plant low-level waste was a liquid waste stream high in nitrates, sodium, and glycolate. It also contained citrate (Agnew 1994b). The tank again had much of its waste volume removed in 1974. Subsequently, ion-exchange waste and evaporator bottoms supernatant were directed to it from tanks in the BY Tank Farm. Tank 241-BX-105 was used as an evaporator feed storage tank from 1976 to 1977 (Anderson 1990). During 1978 the tank became an active receiver of salt well wastes pumped from tanks in the BX and BY Tank Farms (Welty 1988). During this time, the tank also received double-shell slurry feed, and comparatively large transfers of complexed and non-complexed waste from what are now unknown sources.

During its final year of operation (1980) the tank received a large amount of supernatant from tank 241-S-107, as that tank was removed from service. Most of this supernatant was subsequently moved to tank 241-BX-104. At the time of the transfer, tank 241-S-107 contained a variety of wastes, including concentrated REDOX wastes and slurry feed.

Tank 241-BX-105 was deactivated Nov. 17, 1980 (Anderson 1990). The tank was designated interim stabilized after most of the supernatant was removed in March 1981. Supernatant pumping and intrusion prevention were completed in 1986 (Brevick, et al 1994;

Swaney 1994). For a further discussion of interim stabilization and intrusion prevention see the *Tank Characterization Reference Guide* [De Lorenzo et al. 1994].

Much of the solids in tank 241-BX-105 would have settled from wastes added to the tank between 1954 and 1963 (Brevick, et al. 1994). Solids also would have been added from later additions of cladding waste and salt slurry; however, these solids would likely have been removed during the tank's process history. For the *Tank Layer Model* developed by Los Alamos National Laboratory (Agnew et al. 1995), the solids layers presently in the tank are estimated to be unknown (further discussion of the *Tank Layer Model* follows in Section 2.3.2).

The waste transfer history of tank 241-BX-105 is detailed in Table 2-1 and depicted graphically in Figure 2-4. Table 2-2 summarizes the estimated total volumes of specific waste types added to the tank and presents an estimate of the volume of waste types remaining in the tank (from Brevick et al. 1994).

2.3.2 Historical Estimation of Tank Contents

A preliminary estimate of the waste constituents in tank 241-BX-105 has been derived in the *Hanford Tank Content Estimate for the Northeast Quadrant* (Brevick et al. 1994) using a strategy developed at Los Alamos National Laboratory. This strategy employs the *Waste Status and Transaction Summary for the Northeast Quadrant* (Agnew 1994a) to derive an estimate of the solid and liquid layers residing in the tank, the *Tank Layer Model* (Agnew et al. 1995) to define solids layers within the tank, and *Hanford Site Defined Wastes* (Agnew 1994b) to provide chemical and radiochemical definitions for each waste type. Corrections to the original Brevick, et al. (1994) report were made in Engineering Change Notice No. 617835 and these changes have been included in this estimate. The content estimate for tank 241-BX-105 is presented in Table 2-3. An effort is underway to quantify the uncertainty associated with the historical estimates based on tank sampling data (Simpson and McCain 1995).

2.4 SURVEILLANCE INFORMATION

2.4.1 Surface Level Readings

The waste surface level within tank 241-BX-105 is monitored and measured with a FIC gauge through riser 1 (see Figure 2-1). Surface level measurements are taken with the gauge manually on a quarterly basis and input to the Computer Automated Surveillance System (CASS). Since May 7, 1990, the FIC gauge has been set in intrusion mode. Between manual level readings the gauge is poised one inch above the waste surface monitoring the tank should a surface-level increase occur (Brevick, et al. 1994). The most recent manual

Table 2-1. Tank 241-BX-105 Waste Transfer History. (3 pages)

Year	Transaction source or *destination*	Transaction amount, kL	Waste type	Tank total volume, kL (kgal)	Comments
1949	BX-104 *BX-106*	3,810 -1,800	Metal Waste *-Supernatant*	2,010 (530)	Cascade begins filling. Cascade filled in September; began cascading to BX-106[a].
1950 - 1953	BX-104 *BX-106*	4,950 -4,950	Metal Waste *-Supernatant*	2,010 (530)	Cascade filled[a].
1954	Unknown *BX-106* *Unknown*	1,960 -2,010 -1,760	Water[a]. *-Metal Waste* *-Unknown*	204 (54)	Tank sluiced for uranium recovery.
1955	Unknown *Unknown*	223 -201	Unknown *-Unknown*	227 (60)	Leach tank to recover uranium[a].
1956	BY-110 *Unknown*	1,970 -216	UR *-Unknown*	1,980 (524)	Leach tank to recover uranium[a].
1957	Unknown *BC-14 Trench* *BY-102*	19 -1,150 -613	Unknown *-Supernatant* *-Supernatant*	235 (62)	—
1958 - 1962	Unknown Inst. Adjustment	136 11	Unknown —	382 (101)	Surface level instrumentation change added 11 kL (3 kgal) to total waste volume[a].
1963	C-102	1,680	CWP	2.070 (546)	Solids in inventory 401 kL (106 kgal)
1964	C-102 *BX-109* *Unknown*	390 -401 -8	CWP[a] *-Supernatant* *-Unknown*	2,050 (541)	—
1965 - 1967	*Unknown*	-27	*-Unknown*	2,020 (534)	Small unknown losses. Solids in inventory 37 kL (98 kgal).
1968	BX-104 *BY-103* C-102	1,650 -2,840 -1,230	IX/CW *-Supernatant* CWP	2,070 (546)	—
1969 - 1971	Unknown *Unknown*	42 -129	Unknown *-Unknown*	1,980 (523)	Unknown gains and losses.
1972	*BX-103* *BX-106* Unknown *Unknown*	-1,190 -503 11 -83	*-Supernatant* *-Supernatant* Flush Water[a] *-Unknown*	212 (56)	—

Table 2-1. Tank 241-BX-105 Waste Transfer History. (3 pages)

Year	Transaction source or *destination*	Transaction amount, kL	Waste type	Tank total volume, kL (kgal)	Comments
1973	*BX-103*	-1,950	*-Supernatant*		
	BX-104	2,010	BL/IX		
	BX-106	-64	*-Supernatant*	1,580	—
	BX-108	1,640	IX	(481)	
	Unknown	4	Unknown		
	Unknown	-34	*-Unknown*		
1974	*S-110*	-1,440	*-Supernatant*		
	BX-112	1,185	IX/EB	1,580	—
	Unknown	27	Unknown	(417)	
	Unknown	-11	*-Unknown*		
1975	BY-101	76	IX/EB		
	BY-112	121	IX		
	SX-110	-1,070	*-Supernatant*	715	—
	Unknown	8	Water	(189)	
	Unknown	8	Unknown		
	Unknown	-4	*-Unknown*		
1976	BY-112	1,180	IX/EB	1,650	
	BXR-003	34	Unknown	(436)	—
	Unknown	-276	*-Unknown*		
1977	Unknown	136	Unknown	496	Active evaporator feed
	Unknown	-1,290	*-Unknown*	(131)	storage tank[a].
1978	BX-104	212	NCPLX		Active salt well receiver.
	BX-104	-503	*-Supernatant*		Evaporator feed storage.
	BY-101	27	Supernatant		Cross-Site receiver[a].
	BY-102	53	NCPLX		
	BY-111	8	Supernatant	799	Solids in inventory 276 kL
	Cross-Site	367	DSSF	(211)	(73 kgal).
	Unknown	363	NCPLX		
	Unknown	11	Unknown		
	Unknown	-235	*-Unknown*		
1979	*A-102*	-700	*-Supernatant*		Active salt well receiver[a].
	BX-104	-6,17	*-Supernatant*		New photo 3/7/79.
	BY-110	310	CPLX		
	SX-101	4,600	CPLX	424	
	Unknown	3,250	Unknown	(112)	
	Unknown	-1,080	*-Unknown*		
1980	*A-101*	-731	-Supernatant		Solids in inventory 216 kL
	BX-104	231	Supernatant		(57 kgal) 6/30/80[a].
	BX-104	-13,130	-Supernatant		
	S-107	9,390	Supernatant	310	Deactivated 11/17/80.
	SX-101	1,780	Supernatant	(82)	
	Cross-Site	1,870	DSSF		
	Unknown	977	CPLX		
	Unknown	155	DSSF		
	Unknown	-651	NCPLX		

Table 2-1. Tank 241-BX-105 Waste Transfer History. (3 pages)

Year	Transaction source or *destination*	Transaction amount, kL	Waste type	Tank total volume, kL (kgal)	Comments
1981 - 1995	*Unknown* *BX-244*	-57 -61	*-NCPLX* *-NCPLX*	192 (51)	Interim stabilized 3/81. Supernatant pumping completed 8/86[b]. Intrusion prevention completed 9/86.

Note: All information used in the table comes from *Waste Status and Transaction Record Summary for the Northeast Quadrant* (Agnew 1994a) unless otherwise noted.

[a]Anderson 1990.

[b]Welty 1988.

BL = B Plant Low-Level
CPLX = Complexed
CW = Cladding Waste
CWP = Cladding Waste PUREX
DSSF = Double-Shell Slurry Feed
EB = Evaporator Bottoms
X = Ion Exchange
NCPLX = Noncomplexed
UR = Uranium Recovery

Figure 2-4. Tank 241-BX-105 Fill History.

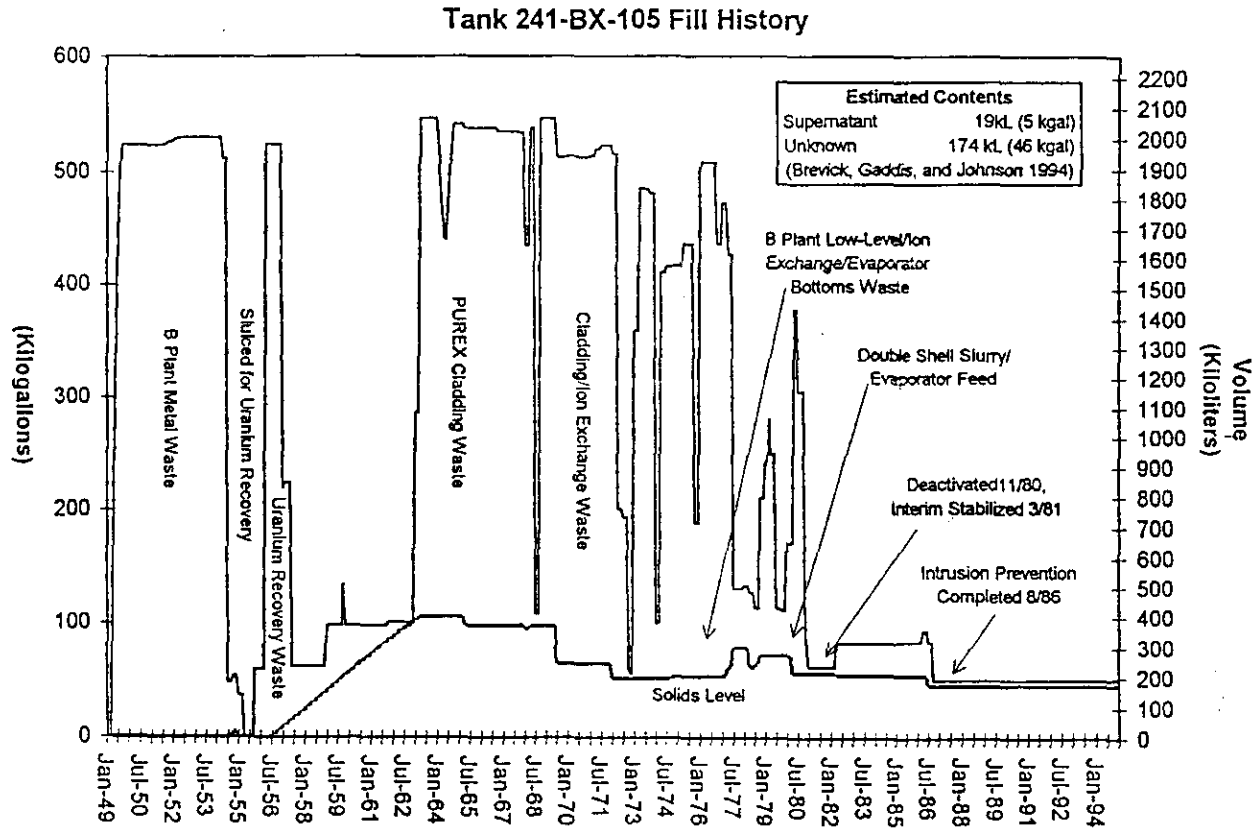


Table 2-2. Estimated Historical and Current Total Volumes of Waste Types Received.*

Waste type	Estimated volume	Years received	Estimated current volume kL (kgal)
B Plant metal waste	8,750 (2,311)	1949 - 53	--
Uranium recovery waste	1,970 (521)	1956	--
PUREX cladding waste	3,300 (873)	1963 - 68	--
Ion exchange/ cladding/B Plant low-level/evaporator bottoms wastes	7,860 (2,077)	1968 - 76	--
Noncomplexed waste	685 (181)	1978 - 80	--
Double-shell slurry feed	2,390 (632)	1980	--
Complexed waste	5,890 (1,556)	1979 - 80	--
Unknown	17,300 (4,571)	1955 - 80	174 (46)
Supernatant	--	--	19 (5)

Agnew 1994a; Brevick, et al. 1994.

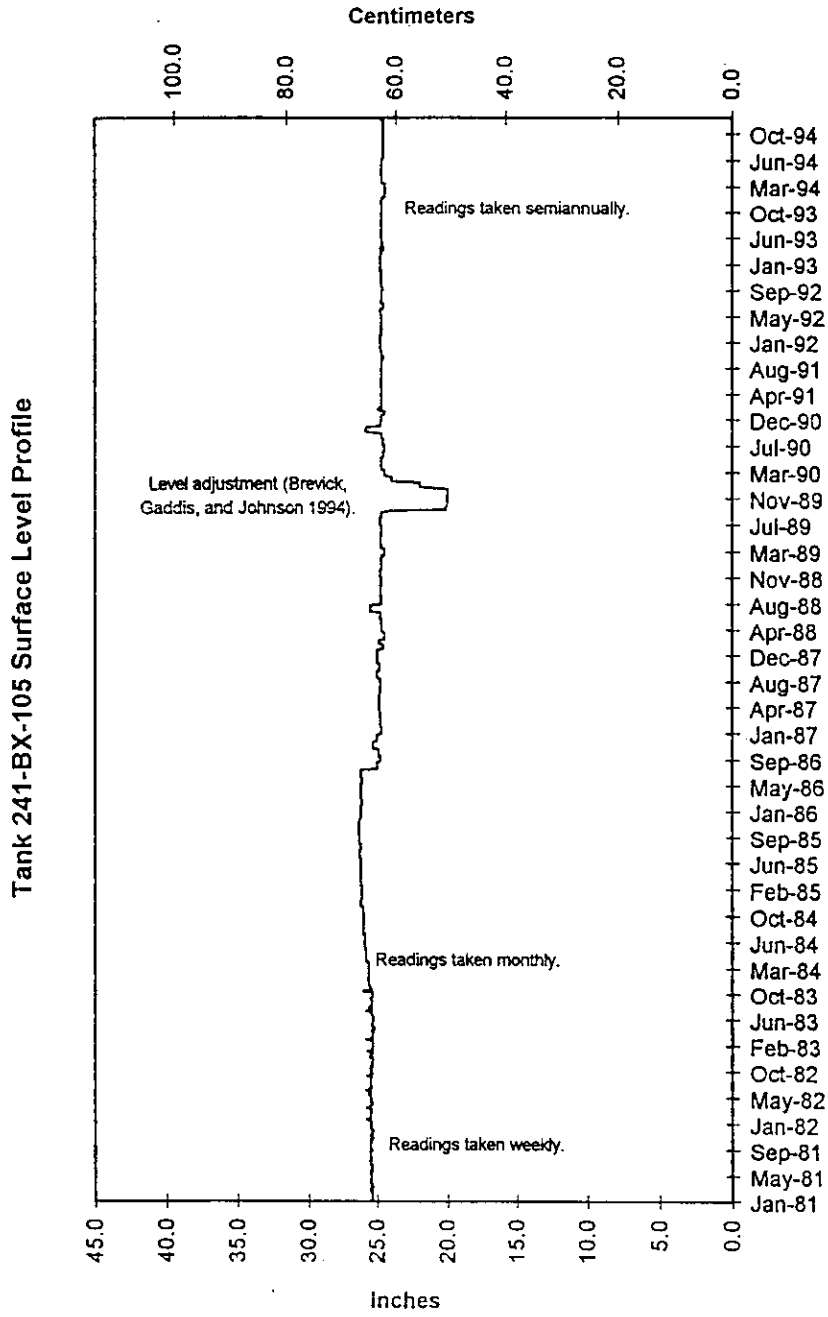
Table 2-3. Tank 241-BX-105 Hanford Tank Content Estimate.*

Physical properties		
Total solid waste	2.62 E+05 kg (46 kgal)	
Heat load	0.112 kW (383 Btu/hr)	
Bulk density	1.51 (g/cc)	
Void fraction	0.486	
Water wt%	57.4	
Total organic carbon wt% C (wet)	0.00	
Chemical constituents		
Analyte	(μg/g)	(kg)
Na^{+1}	1.62 E+05	19,400
Fe^{+3}	11,800	3,100
Ni^{+2}	786	206
Ca^{+2}	1,270	332
OH^{-1}	80,500	21,100
NO_3^{-1}	5,460	1,430
CO_3^{-2}	31,300	8,190
PO_4^{-3}	29,900	7,820
SO_4^{-2}	57,600	15,100
Uranium	1.61 E+05	42,200
Radiological constituents	(μCi/g)	(Ci)
Plutonium	4.59 E-03	1.20 (1.94 E-02 kg)
Cesium	5.98 E-01	157
Strontium	63.2	16,600

*Brevick, et al. 1994.

surface-level reading available from the CASS was 63 cm (24.8 in.) measured April 1, 1995. Available surface-level data from the CASS are plotted in Figure 2-5. Figure 2-5 shows that there have been no significant changes in the surface level since the tank was stabilized in 1981. The level readings indicate a slight rise in surface between 1982 and 1986. This level increase may have occurred due to intrusion. As discussed in Section 2.3.1, 61 kL of

Figure 2-5. Tank 241-BX-105 Surface Level Profile 1981-95.



supernatant was pumped from the tank and intrusion prevention was completed in August of 1986. Figure 2-5 shows only a slight level decrease following supernatant pumping.

Waste levels in the BX farm are measured relative to the bottom of the knuckle which ties the tank wall to the dished tank bottom. 47.3 kL (12.5 kgal) of waste are contained within the dished bottom of the BX tanks below the 0" reference point corresponding to the knuckle bottom (Brevick et al. 1994a). The BX tanks are 22.9 m (75 ft) in diameter. In the cylindrical region of the tank, each centimeter of depth corresponds to a 4.10 kL (1.08 kgal) volume. Based on the 63 cm depth as indicated by CASS, the calculated waste volume is 306 kL [(63 cm)(4.10 kL/cm) + 47.3 kL]. This volume is much higher than the 193 kL historical estimate reported in Hanlon (1995) and Brevick, et al. (1994a). The difference between the historical and FIC indicated waste volumes can only be explained if the surface of the waste is not level across the diameter of the tank. The surface photos (Figure 2-7) seem to confirm a ledge in the waste along the perimeter, with the FIC contacting this ledge. The fact that only a very small drop in waste level was measured at the perimeter following supernatant pumping also seems to verify this assumption.

2.4.2 Internal Tank Temperatures

The single thermocouple tree in tank 241-BX-105 has 14 probes to monitor in-tank temperatures. Thermocouple elevations are not available for this tank (Tran 1993). Thermocouples 1 through 12 have very similar readings from 1974 to 1994. Data for Thermocouples 13 and 14 are available from 1980 to 1987. The data for Thermocouples 13 and 14 are consistent with data from thermocouples 1 through 12. Since May 1994 thermocouples 1, 2, and 7 have been monitored continuously by the Tank Monitoring and Control System with data recorded daily and have continuously trended around 21 °C (70 °F).

Since BX-105 only contains about 63 cm (25 inches) of waste, it is likely that only the bottom one or two thermocouples are measuring actual waste temperatures. The rest are likely measuring tank vapor space temperatures. High temperatures measured by the thermocouples are plotted in Figure 2-6. Figure 2-6 shows that since 1988 the tank temperature has stabilized around 21 °C (70 °F).

2.4.3 In-Tank Photographs

The tank interior was most recently photographed on October 23, 1986. A montage of the photographs are presented in Figure 2-7. The photographs appear hazy with much color variation. Waste surface features cannot be determined from the pictures. Equipment visible in the photographs include a flex and float pump, a submersible pump, a Food Instrument Corporation gauge and some debris (Brevick, et al. 1994). The photos were taken after the tank was supernate pumped in August of 1986. Except for surface drying, the photos should be a good depiction of the current content of the tank.

Figure 2-6. Tank 241-BX-105 Temperature Profile 1984-95.

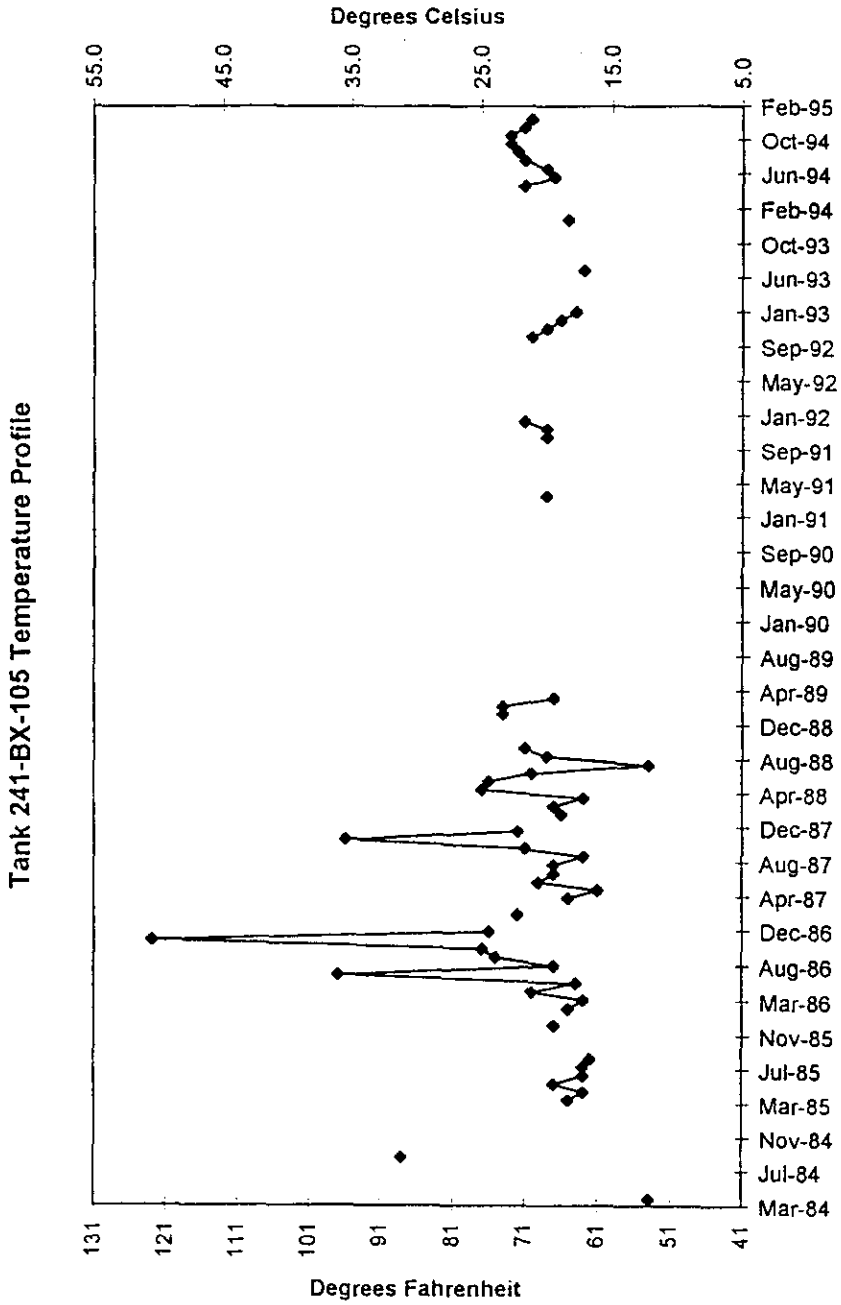


Figure 2-7. Photo Montage of Tank 241-BX-105.

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3.0 TANK SAMPLING OVERVIEW

This section describes two recorded sampling and analysis events for tank 241-BX-105 that occurred after the tank was removed from service in 1980. The first of these sampling events was done in March 1986 when two core samples were taken by push-mode. Extensive chemical and radiochemical analyses were performed on these core samples. In the second event, two 20-in. auger samples were taken in late September and early October of 1994. The auger samples were taken to evaluate the tank contents against safety screening DQO criteria defined in Babad & Redus (1994). Specifically, the auger samples were tested for water content, fuel content, and total alpha content. The results of both sampling events are summarized below.

3.1 DESCRIPTION OF 1986 SAMPLING EVENT

Tank 241-BX-105 was sampled on March 3, 1986 through riser 1 and on March 4, 1986 through riser 8, both samples employing the push-mode core sampling technique. A further description of push-mode core sampling procedures is provided in the *Tank Characterization Reference Guide* (De Lorenzo et al. 1994). Both core samples were taken prior to tank being sealed up to prevent intrusion. According to level instrumentation in the tank, supernatant pumping from the center of the tank after these core samples were taken did not result in a significant drop in waste level below riser 1. The sludge samples taken with the core sampling truck are thus judged to be representative of the current tank contents, which is estimated to contain a very small fraction of supernatant (i.e., the tank contents are currently composed mostly of sludge and salt cake).

The core barrels (samplers) used in core sampling operations are 48.3 cm (19 in.) long. Based on FIC level gauge readings, the core sampling truck operators expected to retrieve 66.8 cm (26.3 in.) of waste in each core sample. Two segments were thus taken from each riser location. The inside diameter of the samplers is 2.22 cm (0.875 in.). Each sampler thus has a usable volume of 187 mL. Normal paraffin hydrocarbon (NPH) was used as a hydrostatic fluid to prevent waste moving up the drill sting as the first samplers were removed.

On the first segment of the core sample taken through riser 1, a very hard layer of waste material was contacted 13.3 cm (5.25 in.) before the end of travel of the drill string. This layer was penetrated with 2.5 cm (1.0 in.) of travel left. The radiation reading on contact through the drill string for the first segment was 350 mrad. On the second (lower) segment taken through riser 1, according to field observations, the tank bottom was contacted 27.6 cm (10.9 in.) inches before the end of drill string travel. The remote latch unit was inadvertently opened as this sampler was being lowered. This sampler was dropped upon retrieval and had no radiation reading through the drill string. Because the drill string contacted the bottom of the tank 27.6 cm (8.1 in.) before the end of drill string travel, the

length of sample expected to be recovered in the first segment was 46.5 cm (18.3 in.). The length expected for the second segment was 20 cm (8.0 in.). From Weiss and Schull (1988), the actual length of sample recovered in the first segment was 36.3 cm (14.3 in.). No sample was recovered for the second segment. The overall recovery for the first core sample (total for both segments) was 54%.

No problems were encountered in taking the first segment for the core sample through riser 8. The radiation reading on contact through the drill string was 500 mrad. During the taking of the second (lower) segment through riser 8, the drill string contacted the bottom of the tank 3.2 cm (1.25 in.) before the end of drill string travel. The sampler did not latch in place and came up with the grapple. The shear pin broke as the sampler was being raised and the sampler dropped to the bottom of the drill string. However, no waste sample was lost. The sampler was retrieved and read 250 mrad on contact through the drill string.

For the second core sample, the first segment was expected to be 24 cm (9.5 in.) long. The second segment was expected to be 42.5 cm (16.8 in.) long. The actual lengths recovered were 34.5 cm (13.6 in.) and 46 cm (18 in.), respectively. The overall recovery for the second core sample was 120 percent.

Table 3-1, below, summarizes the 1986 core sampling event. The 1986 core sampling recoveries indicate a waste depth below risers 1 and 8 of between 56.3 cm (22.3 in.) and 80.5 cm (31.6 in.). The waste depth indicated by the FIC gauge is in this range.

Table 3-1. Tank 241-BX-105, 1986 Sample Data.

Rise number	Segment	Expected length-cm (in.)	Actual recovered length-cm (in.)	Percent recovery	Draill string dose rate-mR/h
R-1	1	46.5 (18.3)	36.3 (14.3)	78	350
R-1	2	20 (8.0)	0	0	No reading
R-8	1	24 (9.5)	34.5 (13.6)	143	500
R-8	2	42.5 (16.8)	46 (18)	108	250

3.1.1 Sample Handling (1986)

The samples were extruded and analyzed and the 222-S Laboratory. According to Weiss and Schull (1988), segment 1 from riser 1 contained 290.97 g of sample, with the following phase distribution:

Solid	105.1 g
Organic	8.2 mL
Aqueous	128.0 mL.

The aqueous phase described by Weiss and Schull was a slurry, consisting of both liquids and solids. The bulk sample is described in Weiss and Schull as being dark brown, with grit-like particles. One poorly formed, greenish crystal with white spots weighing 0.76 g was discovered. The slurry material was separated from the solids phase and centrifuged. After centrifuging, Weiss and Schull is not clear on whether the organic phase was separated off from the aqueous phase before centrifugation. Typically, however, in the 1986 time from any readily observed organic layer was seperated off and discarded prior to analysis on the assumption that it was NPH hydrostatic fluid. The resulting liquid was re-centrifuged. The re-centrifuged supernatant was yellow in color. The re-centrifuged supernatant amounted to 62 mL. This supernatant was bottled and assigned an Analysis ID number of 81XD00XX. The centrifuged solid portion of Sample #811 was combined with the other solids, homogenized, and assigned an analysis ID number of 81XC00XX. Because no sample was recovered in the second segment of this core, the segment 1 sub-samples were treated as composites for the riser 1 core sample.

Segment 1 from riser 8 contained 136.29 g of sample with the following phase distribution:

Solid	40.0 g
Organic	4.5 mL
Aqueous (slurry)	46.3 mL.

The bulk sample was dark brown with dark liquid. Some small hard chunks were observed. The solids and slurry were separated, and the slurry was centrifuged. No explanation is given in the lab report for the apparent discrepancy in the mass balance on this sample.

Segment 2 from riser 8 contained 304.45 g of sample with the following phase distribution:

Solid	294.0 g
Organic	2.0 mL
Aqueous (slurry)	6.5 mL.

The solids were olive green with relatively large quantity of off-white, hard substances which had the consistency of half-set putty and were gummy. The solids were difficult to stir. A couple of slate-like rocks were observed. After stirring, the solids had the appearance of old guacamole with the consistency of cookie dough. It is believed that the slate like rocks were included in the composite, as no mention is made in the laboratory report that they were removed prior to compositing. The solid and slurry phases were again separated, and the slurry phase was centrifuged.

Core composite samples were made using both riser 8 segments. The total quantity of drainable liquid from both samples was combined and yielded 52.8 mL of a bright yellow solution (labeled ID# 82XD00XX). A solids composite sample (82XC00XX) was made by blending segment portions based on wt% fractions of the total recovered core weight. A 12.17 g sample of solids from segment 1 was combined with 89.45 g of solids from segment 2 to form this core composite. After blending the solids mixture was olive green with a gummy peanut butter-like consistency.

3.1.2 Sample Analysis (1986)

The solids and liquids core composite samples were analyzed to determine the concentrations of a wide range of metals, the concentrations of several radioisotopes, the nitrate content, and the total organic carbon (TOC) content. Weight percent water and wt% oxides were also determined, along with bulk density. The reader is referred to Weiss and Schull (1988) for further information on the analytical procedures used. The analytes tested for and the results of the analyses are summarized in Section 4.

The drainable liquor core composites (samples 81XD00XX and 82XD00XX) were filtered and analyzed directly. The solids core composites (samples 81XC00X and 82XC00X) were first washed with water and centrifuged (multiple times). The water was then analyzed for a set of water soluble analytes. The left over (water insoluble) sludge was treated with a 5 M HCL acid solution and centrifuged. Any material not dissolved by the acid was combined with an HCl-HNO_3-HF solution at elevated temperatures in a pressure reactor. The centrifuged liquid solutions from the two acid leaching steps were combined to quantify the water-insoluble analytes.

It should be noted that residual material was left over after the final HCl-HNO_3-HF dissolution step. The amount of insoluble sample material is difficult to quantify because the residual material from each leaching step is weighed wet, and at each step the material can absorb a considerable amount of liquid. Because all the sample material could not be dissolved, the results from the laboratory analyses may not be representative of the full core sample.

3.2 DESCRIPTION OF 1994 SAMPLING EVENT

Tank 241-BX-105 was sampled on September 30, 1994 through riser 2, and October 5, 1994 through riser 6, using the auger sampling method. Riser location can be seen in Figure 2-1. Both of the 50.8 cm (20 in.) auger samples had 19 flutes, with flute one beginning at the auger shaft and flute 19 located at the tip of the auger bit. A further description of auger sampling procedures is provided in the *Tank Characterization Reference Guide* (De Lorenzo et al. 1994). Chain-of-custody forms were completed for each sample. Both auger samples produced a contact dose of 200 mR/h through the drill string.

The auger samples were taken to meet the requirements defined in the *Tank Safety Screening Data Quality Objective* (Babad and Redus 1994). The following subsections describe the sample handling and analysis in the laboratory.

3.2.1 Sample Handling (1994)

The auger sample removed from riser 2 on September 30th, 1994 was transported to the 222-S Laboratory on October 3rd, 1994. The auger sample removed from riser 6 on October 5, 1994 and transported to the 222-S Laboratory on October 6, 1994. Both samples were received and placed in the 1-E2 hot cell for extrusion and analysis by laboratory personnel.

A chain of custody record was kept during the sampling event for both of the samples taken. This document ensures safe transport and maintains a record of personnel involved in sampling and transport of the samples to the laboratory. For a further discussion of chain of custody functions and sample handling information, see the Tank Characterization Plan for 241-BX-105 (Schreiber 1994).

Laboratory observations during extrusion and breakdown for analyses are discussed for each auger sample, separately, in the following paragraphs.

Riser 2 Auger Sample — Extrusion of the riser 2 auger sample took place on October 6, 1994. The riser 2 sample contained a total of 60.0 g of sample. Sample material was present on all 19 flutes of the 20 in. auger. The flutes are numbered such that flute 1 begins at the auger shaft and flute 19 ends at the tip of the auger bit. The spaces between the augers were not filled, rather the sample adhered to the flutes and auger shaft. Although no drainable liquid was present, some creamy, gray, mud-like material fell onto the auger tray during extrusion, and some remained on the tip of the auger. No liner liquid was present, but a portion of the liner was coated with what appeared to be the same material that fell onto the extrusion tray. The visual moisture content in this auger sample was variable. The tray material was runny, while the other material appeared dry. The dry material flaked off the auger when being subsampled.

A 4.90 g subsample (Sample No. S94T000139) was removed from flutes 15 and 16 for DSC/TGA analysis approximately five minutes after extrusion. This sample was grayish white in color and appeared moist with a paste-like consistency. Some flaking occurred during removal. A second subsample (15.10 g) was collected for DSC/TGA analysis from the soft, brown, mud-like material on the auger tray. Some material for archiving was removed from the vial containing this subsample. The vial was later broken during sample loadout from the hot cell. Because the amount of time between breakage and discovery of the break was unknown, it was decided that the remaining sample had been compromised for its intended purpose, and it was discarded. Two other subsamples were taken for safety screening: one from the upper half (flutes 1-9) of the auger, and one from the lower half (flutes 9-19). The last two subsamples were labelled S94T000141/142 and S94T000143/144, respectively, and were collected 30 to 45 minutes after the first subsample (S94T000139).

Riser 6 Auger Sample — The riser 6 auger sample was loaded into the laboratory hot cell and extruded October 7, 1994. No problems were noted while extruding the auger sample. A total of 319 g of sample was collected. Although all flutes of the auger contained some sample, the majority of the sample material was found on flutes 11-15. The flute 11-15 sample material was grayish white, thick, and pasty. Approximately 5 mL of liner liquid was collected but was not retained. Also, a small amount of material fell onto the auger tray and was not retained. The material on all the flutes appeared similar except the material on flutes 11 through 15 which were crusty. This may have been due to sample drying in the hot cell.

The following subsamples were taken from this auger. Sample S94T000146 (5.81 g) was removed for DSC/TGA analysis from flutes 11-15 approximately five minutes after extrusion. Two other subsamples were removed for safety screening analysis from the upper half (flutes 1-10) and lower half (flutes 11-19) of the auger. The subsamples were labelled S94T000148/150 and S94T000147/149, respectively.

Table 3-2 describes and summarizes the sampled taken from both auger samples.

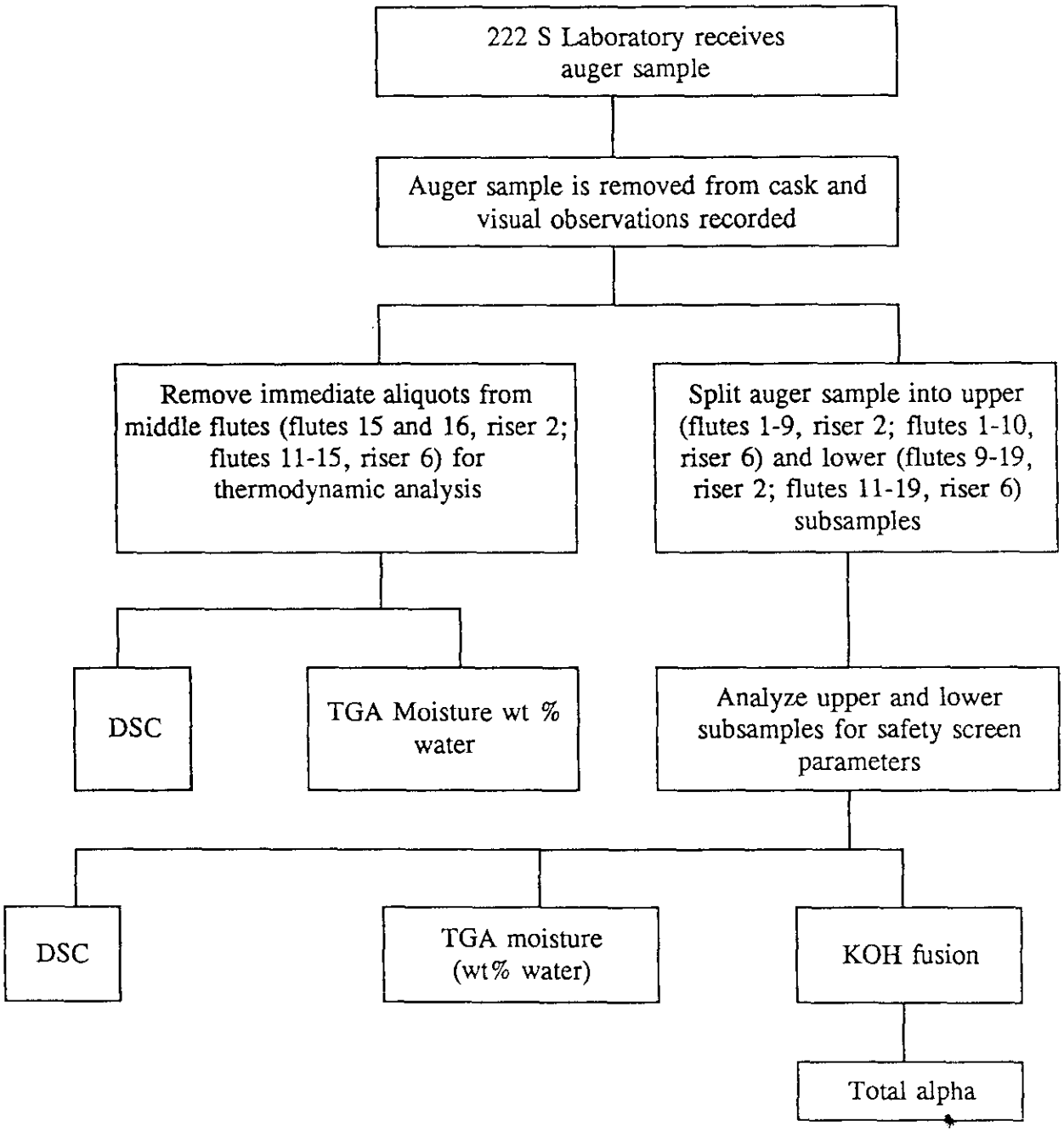
3.2.2 Sample Analysis (1994)

All analyses were performed at the 222-S Laboratory. The analyses performed on the auger samples were limited to those needed to satisfy the safety screening requirements: determination of wt% water by TGA, exothermal activity by DSC, and total alpha content by alpha proportional counting. Additional information on analytical methods can be obtained in the *Tank Characterization Reference Guide* (De Lorenzo et al. 1994). The steps taken to subdivide and analyze the BX-105 auger samples are summarized in the flowchart provided in Figure 3-1.

Table 3-2. 1994 Sample Description Summary for Tank 241-BX-105.

Riser	Sub-sample location	Lab core sample number	Wt. solids (g)	Sample description
2	Flutes 15,16	S94T000139	4.90	Gathered immediately after extrusion; gray, white, moist and pasty
	Auger tray	S94T000140	7.00	Brown, soft and mud-like
	Flutes 1-9	S94T000141/142	2.33	Medium brown and off-white, dry and flaky
	Flutes 9-19	S94T000143/144	5.78	Gray, white, dry and flaky
	Archive (from flutes 15,16)	S94T000151	26.4	Brown, soft and mud-like
6	Flutes 11-15	S94T000146	5.81	Gathered immediately after extrusion; grayish-white, thick and crusty
	Flutes 1-10	S94T000147/149	5.82	Grayish-white, thick and pasty
	Flutes 11-19	S94T000148/150	7.70	Grayish-white, thick and pasty
	Archive	S94T000152	61.7	N/A

Figure 3-1. 1994 Sample Handling and Analysis Flowchart for of Tank 241-BX-105.



The immediate subsamples taken from flutes 15 and 16 from the riser 2 auger sample and flutes 11-15 from the riser 6 sample were analyzed for thermal characteristics by DSC and wt% moisture by TGA. These subsamples were taken as quickly as possible after the auger was removed from the cask to prevent the waste from drying in the hot cell atmosphere.

The upper and lower subsamples from each auger were also analyzed by DSC/TGA for comparison purposes. In addition, the upper and lower subsamples were analyzed for fissile content by total alpha analysis. The DSC and TGA analyses are performed on small (5-20 mg) aliquots of the waste sample.

Before a total alpha analysis could be performed, the waste had to be dissolved. This was accomplished by fusing a solid aliquot (0.2 - 0.5 g) of the homogenized waste in potassium hydroxide and dissolving the flux in hydrochloric acid. The total alpha concentration was determined on a liquid aliquot of the dissolved waste. Each analysis was performed in duplicate with appropriate blanks and standards. The results of these analyses have been reported in the *45-Day Safety Screening for Tank 241-BX-105 Auger Samples, risers 2 and 6* (Bell 1994b) and are summarized in Section 4.

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4.0 ANALYTICAL RESULTS AND WASTE INVENTORY ESTIMATES

This section collects all the data measured for the two sampling events previously described. The primary purpose for the 1986 core sampling event was to compile information necessary for future retrieval efforts. As such, a fairly comprehensive chemical component survey was done on both solid and liquid core samples. The anger samples collected in September and October of 1994 from tank 241-BX-105 were taken to satisfy the *Tank Safety Screening Data Quality Objective* (Babad and Redus 1994). This DQO was developed to allow rapid classification of the tanks containing high-level radioactive waste and to support resolution of various tank safety issues. For tank 241-BX-105, the DQO specifies that the following analyses be performed; TGA to determine water content; DSC to determine fuel energy value, and total alpha analysis to determine the fissile isotope content of the waste.

The results from the 1986 core sampling event and the 1994 auger sampling event are summarized respectively in Sections 4.1 and 4.2. Tank inventory estimates for major analytes, based on the chemical and radiochemical data from the 1986 core sampling event, are provided in Section 4.3.

4.1 ANALYTICAL RESULTS, 1986 CORE SAMPLING EVENT

The 1986 cores sample composites were analyzed for a wide range of metals, several radionuclides, nitrate ion, and TOC. In addition physical measurements were made to determine bulk density and mass loss upon heating (room temperature to 400 °C and from 400 °C to 1,000 °C). The mass loss results are useful for estimating respectively wt% water (free water, interstitial water, and water of hydration) and wt% oxides (Weiss and Schull, 1988). Table A-1 identifies the lab procedures utilized for each analyte analysis. Table 4-1 provides the analytical results for the solid composite samples developed from each core sample. Table 4-2 provides the analytical results for the drainable liquor fractions from each core sample.

The analytical procedures used to determine the analyte concentrations, mass loss as a function of temperature, and sample densities are summarized in Appendix A, Table A-1. The laboratory procedures are described in Bowton and Hiller (1985).

For the solid core composites (Table 4-1), the concentration values reported are the maximum total values reported by Weiss and Schull (1988). The total values are the sum of the concentrations determined for the individual leaching solutions (water, HCl-HNO_3-HF pressure dissolution).

Table 4-1. Solid Core Composite Analyses. (2 sheets)

Component	Unit	Riser 1 composite, segment 1[a]	Riser 8 composite, segments 1 and 2[a]
Aluminum	μg/g	16,800	50,900
Barium	μg/g	2,440	3,950
Bismuth	μg/g	1,290	254
Boron	μg/g	6.09	4.52
Cadmium	μg/g	63.5	25.0
Calcium	μg/g	5,160	7,260
Chromium	μg/g	14,700	2,530
Cobalt	μg/g	1.14	0.00
Copper	μg/g	32.5	22.0
Iron	μg/g	9,800	2,080
Lead	μg/g	831	296
Magnesium	μg/g	2,620	3,900
Manganese	μg/g	2,560	422
Nickel	μg/g	221	73.9
Phosphorus	μg/g	32,000	7,850
Potassium	μg/g	1,430	1,500
Silicon	μg/g	33,300	49,700
Silver	μg/g	39.5	12.1
Sodium	μg/g	142,000	77,800
Strontium	μg/g	70.5	41.9
Zinc	μg/g	206	79.3
Zirconium	μg/g	492	178
Uranium	μg/g	6,600	2,210
Nitrate[b]	μg/g	35,400	31,100
TOC[b]	μg/g	3,760	1,800
pH[c]		> 12	> 12
Mass Loss[c]			

Table 4-1. Solid Core Composite Analyses. (2 sheets)

Component	Unit	Riser 1 composite, segment 1(a)	Riser 8 composite, segments 1 and 2(a)
Room to 400 °C	μg/g	593,000	547,000
400 °C to 1,000 °C	μg/g	64,000	94,800
$^{239/240}$Pu	μCi/g	0.474	0.0847
^{14}C(b)	μCi/g	0.00213	0.00044
^{90}Sr	μCi/g	227	45.9
^{99}Tc	μCi/g	0.0363	0.0306
^{241}Am	μCi/g	1.65	0.336
^{60}Co	μCi/g	0.259	0.0561
^{137}Cs	μCi/g	62.3	45.9
^{129}I(c)	μCi/g	0.000037	0.000023
Bulk density	g/mL	1.58	1.79

(a)sum of water and HCl-HNO_3-HF pressure dissolution fractions.
(b)from water soluble fraction.
(c)analysis on direct sample Weiss and Schull, 1988.

Table 4-2. Drainable Liquid Core Composite Analyses. (2 sheets)

Component	Unit	Riser 1 composite, segment 1	Riser 8 composite, segments 1 and 2
Aluminum	mg/L	5,260	6,590
Barium	mg/L	0.6911	0.8740
Bismuth	mg/L	<12.10	<12.10
Boron	mg/L	13.00	14.40
Cadmium	mg/L	11.40	14.40
Calcium	mg/L	157.0	130.0
Chromium	mg/L	292.0	205.0
Cobalt	mg/L		1.260
Copper	mg/L	7.650	7.040
Iron	mg/L	3.990	4.750
Lead	mg/L	43.30	59.50
Magnesium	mg/L	0.2410	0.5270
Manganese	mg/L	<22.00	<22.00
Nickel	mg/L	87.70	99.60
Phosphorus	mg/L	1,610	1,790
Potassium	mg/L	2,692	2,503
Silicon	mg/L	87.70	191.0
Silver	mg/L	<1.10	<1.10
Sodium	mg/L	128,000	120,300
Strontium	mg/L	1.960	1.630
Zinc	mg/L	3.090	
Zirconium	mg/L	<5.060	<5.060
Uranium	g/L	0.0046	0.00442
Nitrate	M	1.720	2.070
TOC	g/L	9.120	9.750

Table 4-2. Drainable Liquid Core Composite Analyses. (2 sheets)

Component	Unit	Riser 1 composite, segment 1	Riser 8 composite, segments 1 and 2
pH		>12	>12
Mass Loss			
Room to 400 °C	%	68.90	68.50
400 °C to 1,000 °C	%	15.00	14.00
$^{239/240}Pu$	μCi/L	18.90	15.40
^{14}C	μCi/L	0.6320	0.5840
^{90}Sr	μCi/L	11,000	11,500
^{99}Tc	μCi/L	119.0	109.0
^{241}Am	μCi/L	19.40	29.60
^{60}Co	μCi/L	178.0	177.0
^{137}Cs	μCi/L	171,000	179,000
^{129}I	μCi/L	0.046	0.056
Density	g/mL	1.291	1.284

[a] Weiss and Schull 1988.

4.2 ANALYTICAL RESULTS, 1994 AUGER SAMPLE

The 1994 auger samples were collected to evaluate against the safety screening DQO criteria defined in (Babad and Redus 1994). The safety screening data quality objective only requires analysis of wt% water, analysis of fuel content, analysis of thermal output, total alpha analyses for criticality evaluation, and analysis for flammable gas concentration. The safety screening analysis results from the *45 Day Safety Screening for Tank 241-BX-105* (Bell 1994b) are summarized in this section.

Total Alpha Results (1994) — Analyses for total alpha were performed on a fusion digested sample of an alpha proportional counter according to procedure LA-508-101, Rev. D-2. Table 4-3 presents the data results. The table lists the sample numbers and the locations from which the samples were derived. The result column is a simple mean of an

original sample and its duplicate. These results are a specific concentration of the analyte at different sampling points. The riser mean is a simple mean of the listed result. The combined riser mean result is not weighted according to the estimated recoveries of the auger samples. Riser 2 values exceed the 100 nCi/g designation for TRU waste.

DSC/TGA Results (1994) — TGA and DSC were performed on the three different subsamples from each auger sample. The TGA and DSC analyses determine the thermal stability or reactivity of a material. The TGA measures the mass of a sample while the temperature of the sample is increased at a constant rate. Nitrogen is passed over the sample during heating (to prevent oxidation of the sample). Any decrease in the weight of a sample is due to vaporization of a portion of the sample or due to reactions that form gas phase products. The moisture content is estimated by assuming that all TGA sample weight loss, up to a certain temperature (around 150 °C) is due to water vaporization. Typically, the water content of tank waste samples is much greater than the concentration of volatile species, such as short-chain organics. Therefore, the error introduced in making the assumption that all weight loss up to 150 °C is due to water is small. Weight loss due to moisture and other volatile matter can often be differentiated by inflection points in the rate of sample weight loss as well. Weight loss at higher temperatures (e.g., above 200 °C) is attributed to the evolution of reaction product gases. Oxidation of organics in the sample by $NaNO_3$, for example, would release carbon dioxide.

Table 4-3. Total Alpha Results from 1994 Auger Sample.*

Sample number	Sample location	Sample identification	Result	Riser mean	Combined mean
			(μCi/g)	(μCi/g)	(μCi/g)
94T000142	Riser 2, upper half of auger	Flutes 1 - 9	0.589	0.430	0.221
94T000144	Riser 2, lower half of auger	Flutes 9 - 19	0.270		
94T000149	Riser 6, upper half of auger	Flutes 1 - 10	0.0120	0.0115	
94T000150	Riser 6, lower half of auger	Flutes 11 - 19	0.0110		

*Bell 1994b.

The wt% water by TGA was performed using laboratory procedure LA-560-112, Rev. A-2. The analyses were run in duplicate. The results of the TGA analyses are summarized in Table 4-4. The first transition is complete between 100 °C and 140 °C. As discussed above, in this region, endotherms are mainly attributed to the loss of bulk and interstitial water. The second transition occurred between 190 °C and 490 °C. The phenomena demonstrated in this region could be attributed to the loss of covalently bound water molecules (e.g., dehydration of aluminum or hydroxide [$Al(OH)_3$]) or the release of gaseous reaction products.

DSC measures the heat input necessary to keep a sample and a reference substance isothermal as the temperature is increased linearly. DSC can determine the onset temperature for exothermic or endothermic reactions and can quantify the heat of reaction.

DSC analyses were performed under a nitrogen atmosphere using procedure LA-514-113, Rev. B-1. The DSC analyses were performed in duplicate (runs 1 and 2). Like the TGA analysis, the DSC scans showed two distinct transition areas. The enthalpy changes were all positive. Positive enthalpy changes indicate endothermic reactions. The temperature range, temperature at maximum enthalpy change, and the magnitude of the enthalpy change (J/g on a wet weight basis) are provided for both transitions in Table 4-5. The first transition represents the endothermic reaction for the evaporation of the free and interstitial water. The second endothermic transition probably represents the energy (heat) required to remove the bound water from hydrated compounds such as aluminum hydroxide or to melt salts such as sodium nitrate. No exotherms were observed for any of the samples run. The samples were run up to 450 °C. Exotherms above 450 °C are generally not a safety concern, as it is unlikely that the tank contents could ever be heated to such elevated temperatures.

4.3 PROJECTED TANK INVENTORIES

Chemical and radiochemical inventory estimates for Tank BX-105 are derived from the 1986 core sample results. The solids composite and liquid composite results from both cores are used to estimate the projected tank inventory for each analyte. The 1986 core samples results are judged to be valid to estimate the current contents of the tank, because the

Table 4-4. Thermogravimetric Analysis Results for Tank 241-BX-105.*

Subsample location	Laboratory core sample number	Run	Transition 1 (30 °C to 140 °C)		Transition 2 (190 °C - 490 °C)	
			Weight loss (%)	Mean	Weight loss (%)	Mean
Riser 2 - flutes 15,16	S94-T000139	1	9.86	10.7	26.4	25.9
		2	11.5		25.4	
Riser 2 - flutes 1 through 9	S94-T000141	1	13.8	10.5	17.3	19.7
		2	7.24		22.0	
Riser 2 - flutes 9 through 19	S94-T000143	1	15.7	14.6	20.3	20.6
		2	13.4		20.9	
Riser 6 - flutes 11 through 15	S94-T000146	1	19.0	18.9	21.6	21.8
		2	18.8		22.0	
Riser 6 - flutes 1 through 10	S94-T000147	1	4.91	5.53	29.2	28.6
		2	5.54		28.0	
Riser 6 - flutes 11 through 19	S94-T000148	1	16.2	15.5	23.6	23.6
		2	14.8		23.5	

*Bell 1994.

Table 4-5. Differential Scanning Calorimetry Energetic Results for Tank 241-BX-105.

Subsample location	Laboratory core sample number	Run	Transition 1			Transition 2		
			Temp range (°C)	Peak temp (°C)	Delta H (J/g)	Temp range (°C)	Peak temp. (°C)	Delta H (J/g)
Riser 2 - flutes 15, 16	S94T000139	1	40 - 140	118	460	210 - 320	283	387
		2		116	498		291	517
Riser 2 - flutes 1 to 9	S94T000141	1	50 - 160	117	330	200 - 340	289	433
		2		126	465		284	570
Riser 2 - flutes 9 to 19	S94T000143	1	40 - 160	131	488	200 - 330	289	472
		2		110	524		289	626
Riser 6 - flutes 11 to 15	S94T000146	1	40 - 160	122	508	210 - 310	284	441
		2		123	547		286	470
Riser 6 - flutes 1 to 10	S94T000147	1	40 - 140	106	217	200 - 300	287	256
		2		108	732		285	742
Riser 6 - flutes 11 to 19	S94T000148	1	40 - 160	119	661	180 - 320	285	694
		2		109	667		285	670

Bell 1994.

samples were taken after the tank was taken out of service (in 1980) and interim stabilized (in 1981). Approximately 60 kL (16 kgal) of supernatant were pumped from the tank shortly after the core samples were taken. This supernatant, however, was contained in a pool in the center of the waste. The sampled waste from the perimeter of the tank, below risers 1 and 8, did not contain much surface liquor based on in-tank photographs. The fact that the FIC gauge did not drop significantly is further evidence that the waste near the perimeter did not support much supernatant material. Because the tank may have dried due to evaporation, since 1986, the projected inventories for the liquid phase of the waste may be biased high. With the exception of the water soluble species ^{60}Co, ^{99}Tc, ^{129}I, ^{139}Cs, most of the analytes of concern are contained in much higher quantities in the solids (sludge) phase of the waste.

The average of the analyte concentrations determined for the solids composites from both cores is multiplied by the total estimated mass of solids in the tank to determine analyte inventories for the solids phase. Analyte concentrations in the liquids phase are determined by multiplying the average of the analyte concentrations determined for the liquids composites from both cores by the total estimated liquid volume of the tank. Total analyte inventories for the tank can be determined by simply adding the inventories of both phases.

The volume of liquid in the tank is estimated from the weights and volumes of the centrifuged solids and liquids phases determined by Weiss and Schull (1988) for both core samples. For the first segment core sample taken from riser 1, Weiss and Schull report that 62 mL of liquid was recovered after centrifuging. Weiss and Schull also estimate that 36.3 cm (14.3) inches of waste was recovered in segment 1. The sampler is 48 cm (19 in.) long and can contain a total of 187 mL of waste (De Lorenzo et al. 1994). The 48 cm segment therefore corresponds to a total estimated volume of 140 mL [(36.3 cm)(187 mL)/(48 cm)]. Based on the riser 1 results, the volume fraction of liquid in the waste is 0.44 [(62 mL)/(140 mL)]. For the core sample from riser 8, Weiss and Schull (1988) recovered a total of 52.8 mL of drainable liquid, after centrifuging, from both core segments. The total estimated length of waste in both segments was estimated to be 80 cm (31 in.), for a total volume of 305 mL. Based on the riser 8 core sample, the volume fraction of liquid in the waste is 0.17 [(52.8 mL)/(305 mL)]. The quantity of liquid separated from the riser 1 core is greater than the total liquid recovered from the much larger volume of sample taken from riser 8. This indicates that the waste is wetter under riser 1 than riser 8. It is difficult to quantify just how much wetter, because a full core sample was not recovered from riser 1. For lack of more complete characterization data, the average of the two liquid volumes is used to determine the liquid volume for the entire tank. The estimated total waste volume for the tank is 193 kL (51 kgal) (Hanlon 1995). The estimated volume of drainable liquid in the tank is thus estimated to be:

$$\text{Liquid volume} = (193\ \text{kL})(0.44 + 0.17)/2 = 58.9\ \text{kL}.$$

The estimated volume of solids in the tank is 134 kL (193 kL - 58.9 kL). The bulk densities of the solids composites from both cores was found to be respectively 1.58 g/mL and 1.79 g/mL (see Table 4-2). Using an average solids density, the estimated total mass of solids in the tank can be calculated as:

Solids mass = (134,000 L)(1.58 kg/L + 1.79 kg/L)/2 = 226,000 kg.

The analyte inventories by phase, for the tank are summarized in Table 4-6. Columns 2 and 4 provide the average solids and liquids composite concentrations, as determined from the data for each core from Tables 4-2 and 4-3. Columns 3 and 5 provide the inventory estimates for the solids and liquids phases respectively. The solids inventory was calculated by multiplying the solids concentrations by the solids mass determined above. The liquids inventory was calculated by multiplying the average liquids concentration by the liquid volume determined above. Total analyte inventories for the tank can be determined by simply adding the inventories for both phases.

It should be noted that the inventory estimates in Table 4-6 contain a large degree of uncertainty. They are based on analytical results from a partial core sample taken from near the perimeter of the tank to one side (the lower 20 cm of waste was not sampled in riser 1) and a full core sample was taken from near the perimeter on the other side. Each core sample only represents a 2.2 cm in diameter plug out of a 23-m diameter tank. Since the tank has a dished bottom, analytical information regarding the lowest 47 kL of waste is not available from either core sample. As discussed above, estimating the liquid content of the waste is difficult because supernatant was pumped from the tank after the core samples were taken and because evaporative drying has likely occurred since the core samples were taken. Based on the core sample results, there appears to be lateral heterogeneity in the waste (see Section 5.3). Verticle heterogeneity is likely as well, based on the operating history of the tank. The waste near the center of the tank may not be similar to the waste sampled near the perimeter (e.g., waste in center is likely to be wetter than waste near the perimeter, based on in-tank photographs). Because of all the uncertainties involved in extrapolating the inventories from core sample results, statistical confidence limits cannot be determined for the inventories.

The inventory values reported in Table 4-6 are for informational purposes only. No conclusions regarding safety screening issues are drawn in this report solely or primarily from the 1986 data. Qualitatively, the 1986 core sample results indicate that bismuth from the early metal waste heel is present. The high concentrations of uranium, sodium, and phosphorus (in the assumed form of PO_4) in the tank are consistent with the receipt of metal waste and uranium recovery waste early in the tank's operating history. The high concentrations of aluminum and sodium are consistent with the receipt of PUREX cladding waste from 1963 through 1968 and with the receipt of REDOX waste supernatant from tank 241-S-107 in 1980. The high ^{137}Cs content is likely due to the receipt of IX waste between 1973 and 1976. The TOC concentration in the liquids phase is quite high. This could be

due to contamination by the NPH hydrostatic fluid used in core sampling. Weiss and Schull (1988) do not indicate if the organic layer observed upon extrusion was discarded prior to analysis. Typically, in 1986 organic phases were sperated off and discarded prior to analysis.

Table 4-6. Projected Analyte Inventories, By Phase, for Tank BX-105.

Analyte	Mean solids concentration	Projected solids inventory	Mean liquids concentration	Prejected liquids inventory
Metals	(μg/g)	(kg)	(mg/L)	(kg)
Aluminum	33,900	7,660	5,930	349
Bismuth	3,200	723	0.783	4.6 E-02
Calcium	6,210	1,400	144	8.48
Chromium	8,620	1,950	249	14.7
Iron	5,940	1,340	4.37	0.257
Sodium	110,000	24,900	124,000	7,300
Silicon	41,500	9,380	139	8.19
Uranium	4,410	997	4.51 E-03	2.66 E-04
Organics	(μg C/g)[a]	(kg)	(g/L)	(kg)
Total Organic Carbon	2,780	628	9.44	556
Anions	(μg/g)	(kg)	(mg/L)	(kg)
NO_3^-	33,300	7,530	1.90	0.112
PO_4^{-b}	61,000	13,800	5,210	307
Radionuclides	(μCi/g)	(Ci)	(μCi/L)	(Ci)
^{241}Am	0.993	224	24.5	1.44
^{14}C	1.29 E-03	0.292	0.608	3.58 E-02
^{60}Co	0.158	35.7	177	10.4
^{137}Cs	54.1	12,200	1.75 E+05	10,300
^{129}I	3.00 E-05	6.78 E-03	5.10 E-02	3.00 E-3
$^{239/240}Pu$	0.279	63.1	17.1	1.01
^{90}Sr	136	30,700	11,200	660
^{99}Tc	3.35 E-02	7.57	114	6.71

[a]micrograms of carbon per gram of waste.
[b]Conversion from ICP phosphorus results.

5.0 INTERPRETATION OF CHARACTERIZATION RESULTS

The purpose of this section is to evaluate the overall quality and consistency of the available results and to assess and compare these results against historical information and program requirements.

5.1 ASSESSMENT OF SAMPLING AND ANALYTICAL RESULTS

This section evaluates sampling and analysis factors that may impact interpretation of the data. These factors are used to assess the overall quality and consistency of the data and to identify any limitations in the use of the data.

5.1.1 Field/Laboratory Observations

The failure to recover any sample from the second segment from Riser 1 in the 1986 sampling event means the composite data from the Riser 1 and Riser 8 core samples can not be compared. The small amount of waste material recovered in the 1994 auger sample through Riser 2 reduces the confidence in the representativeness of that sample as well.

NPH was used as a hydrostatic fluid in the 1986 sampling event. an organic phase was observed upon extruding each core segment. the laboratory report does not indicate if this organic phase was separated off from the rest of the sample material prior to analysis. If not separated, the presence of NPH fluid in the sample material may have biased the TOC results on the 1986 samples high.

5.1.2 Quality Control Assessment of Data

The data report (Weiss and Schull 1988) for the 1986 sampling event does not include quality control information. Quality control tests (standards, spikes, and duplicates) were not used as extensively in 1986 as is the current practice.

Appropriate standards, spikes, and duplicates for quality control of the 1994 auger sample analyses were performed and are summarized in Table 5-1. Standards are used to estimate the accuracy of the analytical method, and are evaluated prior to and concurrent with sample analysis. Standards contain the analytes of interest at known concentrations. Standard solutions may or may not be independent of the standard used for calibration. The criterion for standard recovery is 100 $\pm$10 percent. If a standard is above or below the criterion, then the analytical results may be biased high or low, respectively. As can be seen in Table 5-1, all standard recoveries for percent water, DSC, and total alpha were well within limits.

Table 5-1. Quality Control Summary, 1994 Auger Samples.

Riser	Analyte	RPD %[a]	Standard %[b]	Spike rec. %[b]
2	% Water (lower half, immediate sample)	15.6	99.50	NA
	% Water (upper half)	62.1	98.23	NA
	% Water (lower half)	15.8	98.87	NA
	DSC (lower half, immediate sample)	NA	100.9	NA
	DSC (upper half)	NA	100.9	NA
	DSC (lower half)	NA	100.2	NA
	Total alpha (upper half)	2.21	97.90	77.80
	Total alpha (lower half)	49.6	97.90	70.10
6	% water (lower half, immediate sample)	1.06	99.50	NA
	% water (upper half)	12.1	97.40	NA
	% water (lower half)	9.04	99.00	NA
	DSC (lower half, immediate sample)	NA	100.9	NA
	DSC (upper half)	NA	100.2	NA
	DSC (lower half)	NA	104.0	NA
	Alpha (upper half)	1.72	96.33	84.10
	Alpha (lower half)	30.7	96.33	NR

[a]10% limit.
[b]Range = 90-110%.
NA = not applicable or available.
NR = not reported.
RPD = relative percent difference.
DSC = differential scanning calorimetry.

Matrix spikes are used to estimate the bias of the analytical method due to matrix interferences. Spike samples are prepared by splitting a sample into two aliquots and adding a known amount of a particular analyte to one aliquot to calculate a percent recovery. The quality control criterion for matrix spikes is 100 $\pm$20 percent recovery. Spikes were only conducted on total alpha. As can be seen from Table 5-1, most all the spike recoveries were

below the quality control limits. This indicates that the total alpha results may be biased low. The low spike values are most likely caused by absorption of alpha particles by residual solids on the mount. This bias is not considered significant because the total alpha results are more than 70 times lower than the DQO criteria (see Section 5.5.1).

Duplicates are used to measure analytical precision and the homogeneity of laboratory samples. A relative percent difference (RPD) is calculated for each duplicate result and is reported in Table 5-1. The RPD is defined as the absolute difference between duplicate measurements, divided by the mean. The total alpha RPD for the samples from the lower auger flutes from both risers exceeded the desired 10 percent precision for safety screening. The counting error for the samples from lower auger flutes from riser 2 was 9 percent (Bell 1994b). The 49 percent RPD for the lower auger samples from riser 2 was therefore probably the result of: (1) analytical error, (2) excessive solids on the mount, or (3) sample heterogeneity. The counting error for the samples from the lower auger flutes from riser 6 was 34.6 percent (Bell 1994), because of the low concentration of alpha present in the sample. The large RPD for the lower riser 6 samples can be accounted for by limited sensitivity of the instrument used to detect total alpha. The large RPDs for the lower auger sample total alpha results are not considered to be significant because the highest total alpha result is nearly 100 times below the safety screening criteria.

Four of the six wt% water results had RPDs greater than 10 percent. The RPD for the upper portion of the riser 2 sample was 62.1 percent. This large RPD was probably caused by sample heterogeneity. Visual observations of the auger sample indicate that the moisture content was variable, even on adjacent flutes. Visual observations from the 1986 riser 1 core sample extrusion (riser 1 is located adjacent to riser 2) indicated the presence of large crystals and different colored solids phases. The small quantity recovered from riser 2 and the small subsample taken (2.3 g) also increases the potential for drying during handling, which can lead to larger variations in the results.

5.1.3 Data Consistency Checks

The ability to assess the overall consistency of the 1986 cores samples and the 1994 auger samples is limited because of sampling anomalies and the limited number of analyses performed at different depths. None of the samples were analyzed for the same component by more than one method. Therefore, no consistency checks can be made based on different analysis methods. There is insufficient anion data from the 1986 sampling event to perform an accurate mass and charge balance on the 1986 results. No anion or metal data are available from the 1994 auger sampling event.

For auger samples, sample drying during extrusion and handling is a concern. For the riser 2 auger sample, the subsample removed shortly after extrusion (139) has a moisture content 3.9 percentage points lower than the other subsample removed from the lower half of the auger 40 to 45 minutes later (143). However, the TGA scan for sample 139 was integrated to approximately 90 °C, while the 143 sample was integrated to

approximately 200 °C (Bell 1994). Therefore the percent moisture for the sample (139) is probably biased low relative to 143. Results for auger samples taken from riser 6 also support the conclusion that extreme sample drying during extrusion and handling did not occur. Sample 148 from the lower half of the auger contained an average of 15.5 percent water, while the sample taken 5 minutes after extrusion from the lower half of the auger contained an average of 18.9 percent water.

5.2 COMPARISON OF RESULTS FROM DIFFERENT SAMPLING EVENTS

There are only two sets of data that are comparable between the 1986 core sampling event and the 1994 auger sampling event. Weight losses over two different temperature ranges were measured on the 1986 samples: room temperature to 400 °C, and 400 °C to 1,000 °C. The weight loss results for the core composites up to 400 °C can be compared against the TGA results from the 1994 auger samples, which were integrated up to 450 °C. Also, the total alpha results from the 1994 auger sampling event can be compared against the combined concentration estimates for $^{239/240}$Pu and ^{241}Am from the 1986 core sample event.

5.2.1 Comparison of Core Sample Weight Loss and Auger Sample TGA Results

The auger sampling device is only effective at recovering solids and sludge materials. Only the solids composite weight loss results from the 1986 core sample are therefore compared against the auger sample TGA results. This comparison is provided in Table 5-2. For the 1994 auger samples, the weight loss reported in the table is the average of the three mean TGA results (lower half immediate sample, lower half, and upper half) from Table 4-4. The auger sample TGA results were averaged for comparison because the solids from each core sample were composited prior to analysis.

Table 5-2. Comparison of Core Sample and Auger Sample Thermogravimetric Analysis Results.

Sample/source	Temperature range	Weight loss
1986 solids composite/ riser 1	Room temp to 400 °C	59.3%
1986 solids composite/ riser 8	Room temp to 400 °C	54.7%
1994 auger samples/riser 2	30 to 490 °C	34.0%*
1994 auger samples/riser 6	30 to 490 °C	38.0%*

*Average of mean TGA results from upper, lower and lower immediate auger samples.

Table 5-2 shows that auger sample results are lower than the core sample results. This is not unexpected for a couple of reasons: 1) Tank BX-105 was supernatant pumped (6.1×10^4 L) after the 1986 core samples were taken, and 2) the auger samples were taken 3 years after the core samples. Over time, the waste could be expected to lose moisture due to evaporation. The riser 2 auger sample recovery was poor, making tests results on that sample especially vulnerable to heterogeneity effects. The subsamples taken for analyses on both auger samples were very small, compounding this problem.

5.2.2 Comparison of $^{239/240}Pu$ and ^{241}Am Results with Auger Total Alpha Results

Table 5-3 compares the gross alpha activities determined on the 1994 auger samples with the sum of the individual alpha emitters as measured on the 1986 core solids composites. The activity concentrations of the individual alpha emitters from the core samples were summed as follows for comparison against the 1994 gross alpha results:

$$\text{Total Alpha Activity} = {}^{141}\text{Am Activity} + {}^{239/240}\text{Pu Activity}$$

Table 5-3. Comparison of Auger and Core Sample Total Alpha Activities.

Location/sample	Method	Total alpha activity (μCi/g)
Riser 1, solids composite (segment 1)	Sum of alpha emitters ($^{239/240}Pu + {}^{241}Am$)	2.12
Riser 2, auger samples	Gross alpha result	0.27 (upper half)
		0.59 (lower half)
Riser 8 solids composite (segments 1 and 2)	Sum of alpha emitters ($^{239/240}Pu + {}^{241}Am$)	0.42
Riser 6, auger samples	Gross alpha result	0.011 (upper half)
		0.012 (lower half)

The total alpha results for the riser 6 auger sample are significantly lower than the total alpha concentrations determined for the riser 8 core composite. Risers 8 and 6 are located adjacent to each other on the same side of the tank. Although the variability in the total alpha data cannot be explained, all of the data are well below the safety screening criteria.

5.3 TANK WASTE PROFILE

The 1994 auger sample TGA results indicate that the moisture content of the waste is higher at the bottom of the waste than at the top. This distribution of moisture could be expected if the supernatant had evaporated from the surface of the waste. Photographs taken inside the tank in 1986, shortly after supernatant pumping, show a relatively dry ring of waste around the edge, with wetter waste in the middle. The tank waste dried out for 8 years before it was auger sampled. The auger samples were taken from risers near the edge of the tank above the dried out ring.

Because the 1986 core samples were composited prior to analysis, vertical distribution of components within the waste cannot be evaluated. The 1986 data appear to show some significant differences for the riser 1 and riser 8 components. Aluminum appears to be in much higher concentrations under riser 8; whereas, sodium, bismuth, chromium, iron, manganese, phosphorus, uranium, $^{239/240}Pu$, ^{90}Sr, ^{241}Am and ^{60}Co are in significantly higher concentrations under riser 1 (see Table 4-2). Although the bottom segment from the riser 1 core was not recovered, (corresponding to the lower 20 cm of waste [out of 63 cm estimated depth]), it can be concluded that the differences in the riser 1 and 8 results are not due to vertical heterogeneity alone. For instance, if the assumption is made that the tank is horizontally homogeneous, and that all the sodium, manganese, ^{241}Am, etc. is contained only in the top 43 cm of waste (corresponding roughly to the depth of the riser 1 segment), the concentrations in the riser 8 composite would be expected to be diluted only by a factor of about one-third (20 cm/63 cm). The riser 8 results for the cations and radioisotopes listed above differ from the riser 1 results by more than this amount, indicating that there is horizontal variability in the tank waste. As discussed in Section 3.1, there was considerably more drainable liquid in the single segment taken from riser 1 than in both segments taken from riser 8. This also indicates lateral heterogeneity.

The total alpha results for the 1994 auger samples are difficult to evaluate for vertical distribution effects due to problems encountered in performing the tests (e.g., high RPDs on both lower auger samples, low spike recovery on all samples).

5.4 COMPARISON OF TRANSFER HISTORY AND ANALYTICAL INFORMATION

Physical properties and historical analyte inventory estimates, based on the Tank Layer Model, are compared with applicable analytical data from the 1986 and 1994 sampling events in Table 5-4. The historical estimates in the second column of the table are reproduced from Brevick et al. (1994) and are the same values reported in Table 2-3. The estimates in column 2 exclude the supernatant layer of the tank, estimated in Brevick to contain 19 kL (5 kgal) of liquid waste.

The analytical estimates for the 1986 sampling event are shown in column three and are based on the analysis in Section 4.3 and the values reported in Table 4-6. To estimate the total mass of the separable liquid phase in the tank, the liquids volume reported in Section 4.3 was multiplied by the average of the densities determined for the riser 1 and riser 8 liquids composites. The total heat load estimate in column 3 is the sum of the heat load contributions from each of the radionuclide inventories in column 3. Table B-1 in the appendix provides the decay heat generation rates for each radionuclide and summarizes the calculations for determining the column 3 heat load value. The wt% water reported in column 3 is based on the temperature induced sample weight losses measured by Weiss and Schull (1988) for riser 1 and riser 8 solids and liquids composites. Only the weight loss results for the room temperature to 400 °C range were assumed to reflect water content. The average weight loss from the two solids composite results was combined with the average weight loss from the two liquid composite results, according to the weight fraction of each phase, to yield to the overall wt% water for both phases. This estimate is probably biased on the high side because some of the weight loss at elevated temperatures may be due to redox reactions within the waste which release product gases. The phosphate estimate reported in column 3 was determined from the phosphorus result reported in Table 4-6.

The wt% water reported for the 1994 auger sample in column 4 is the average of the mean weight loss values reported in Table 4-4 for the first transition 1 (30 °C to 140 °C). For DQO screening purposes, only this first transition can be credited in estimating water content. The value is not comparable to the 1986 value which is based on heating of samples all the way to 400 °C. The mean total alpha value from Table 5-3 was multiplied by the estimated solids mass of 3.58×10^5 kg from Section 4.3 to determine the $^{239/240}$Pu value for column 4. The $^{239/240}$Pu value for column 4 includes other alpha emitting isotopes besides $^{239/240}$Pu, such as ^{241}Am. It is thus biased on the high side.

The total waste mass estimate for the 1986 sampling event is based on the total volume estimate from Hanlon (1994), the drainable liquids content of the core samples, and the measured bulk densities of the solids and liquids composites from the core samples. The historical total waste mass estimate is also based on the Hanlon volume estimate but excludes the supernatant layer. The historical and analytical waste mass estimates are comparable, given that the supernatant layer is not included in the historical estimate.

The historical and analytically determined bulk densities for the waste agree well. Water wt% is also very comparable between the Brevick and 1986 analytical estimates. The wt% water estimate from the 1994 auger samples is significantly lower than the other two estimates. The primary reason for the discrepancy between the 1986 and 1994 results may be that the tests are not directly comparable. The 1986 samples were heated to 400 °C. The 1994 results correspond to weight loss measured only up to 140 °C. The 1986 results are expected to be significantly higher due to loss of water of hydration, which occurs above 140 °C, and release of gaseous reaction products. As discussed in Section 5.2, part of the difference in the 1986 and 1994 weight loss results may be due to evaporative drying of the tank contents over the eight years between sampling events. Another reason the 1994 results

might be expected to be lower is that the auger sampling device is not as effective at recovering wet samples (slurries) as the core sampling device. Thus, the auger samples would be expected to be biased on the low side with respect to the entire tank contents. The true water content of the waste is probably between the 1986 and 1994 analytical estimates in Table 5-4.

Table 5-4. Comparison of Historical and Analytical Tank Content Estimates.

Physical property or constituent	Historical estimate, Brevick (1994)[a]	Analytical estimate, 1986 core sample	Analytical estimate, 1994 auger sample
Total waste	2.62 E+05 kg (46 kgal)	3.0 E+05 kg (51 kgal)	ND
Heat load	112 W (383 Btu/hr)	327 W (1120 Btu/hr)	ND
Bulk density	1.51 g/cc	1.56 g/cc	ND
Water wt. %	57.4%	59.7%[b]	12.6%[c]
TOC wt. % (wet)	0%	0.39%	ND
Sodium	19,400 kg	32,100 kg	ND
Iron	3,100 kg	1,340 kg	ND
Nickel	206 kg	38.8 kg	ND
Calcium	332 kg	1,410 kg	ND
NO_3	1,430 kg	7,510 kg	ND
PO_4	7,820 kg	14,100 kg[d]	ND
Uranium	42,200 kg	995 kg	ND
$^{239/240}Pu$	1.2 Ci	64.1 Ci	49.9 Ci[e]
^{137}Cs	157 Ci	22,500 Ci	ND
^{90}Sr	16,600 Ci	31,500 Ci	ND

[a]Historical estimate excludes supernatant layer (5 kgal)
[b]Weight loss up to 400 °C
[c]TGA results integrated to near 200 °C
[d]Estimated from P result, assuming all P forms PO_4
[e]Mean of total alpha results for both risers multiplied by 1986 estimated solids mass of 2.26 E+05 g; includes ^{241}Am
ND = No Data.

According to the waste transfer history (Section 2.3.1, Table 2-1), some organic containing waste was added to the tank (e.g., trace HEDTA in IX waste — 1963 to 1968, glycolate and citrate in B Plant low level waste — 1972-1976, CMPLX waste — 1979, 1980). Although most of this organic would be expected to have been pumped out during various supernatant transfers, some would be expected to remain in the tank. The 0 wt% TOC historical estimate is thus judged to be unrealistic. On the other hand, the analytical TOC estimate of 0.39 percent may be biased high, because NPH is used as a hydrostatic fluid in core sampling operations, which can contaminate lab samples.

The analytical and historical estimates for sodium, iron, PO_4, and ^{90}Sr agree to within a factor of 3 or 4. The NO_3, uranium, $^{239/240}Pu$ and ^{137}Cs estimates do not agree as well. The analytical results for uranium may be biased low because the core samples did not include the heel material contained in the dish of the tank, where the early metal waste and uranium recovery waste solids would have been expected to collect. This could explain the discrepancy between the analytical and historical uranium estimates. The historical estimate for ^{137}Cs is 2 orders of magnitude lower than the analytical estimate for ^{137}Cs. The results from the 1986 core samples indicate that ^{137}Cs is in very high concentration in the aqueous phase of the waste. This ^{137}Cs probably originated from the IX waste added to the tank between 1968 and 1976.

The discrepancy in the historical and analytical heat load estimates is due to the discrepancies in the radionuclide estimates between the two cases. ^{90}Sr and ^{137}Cs account for most of the heat load in the tank. A statistical comparision between the analytical and historical inventory estimates would not be appropriate as they are based on different models and assumptions.

5.5 EVALUATION OF PROGRAM REQUIREMENTS

Tank BX-105 is classified as a non-Watch List tank; therefore, the only DQO applicable to this tank is the safety screening DQO. This section details the data needs as defined in the *Tank Safety Screening Data Quality Objective* (Babad and Redus 1994), and determines whether Tank BX-105 has been appropriately categorized concerning safety issues. The DQO establishes decision criteria or notification limits for concentrations of analytes of concern. The decision criteria are used to determine if a tank is safe, or if further investigation into the safety of the tank is warranted.

5.5.1 Safety Evaluation

The primary analytical requirements identified in the safety screening DQO are energetics, moisture content, total alpha activity and flammable gas concentration (Babad and Radus 1994).

The waste fuel energy value is determined by DSC. None of the DSC results for the 1994 auger samples exhibited an exotherm. TOC and cyanide results are other indicators of waste fuel content. Centrifuged sludge (solids) results from the 1986 core samples indicate that the TOC levels are relatively low (1,800 to 3,700 μg/g). The TOC in the drainable liquid from this sampling event was high (9.75 g/L). It is possible that NPH hydrostatic fluid contamination could be the cause of these high values. It is also possible that the TOC method used did not adequately eliminate carbonate interference. Using both the centrifuged sludge and drainable liquid analyses from the 1986 core sampling event the TOC content for the entire tank contents is estimated to be 0.39% wet weight (see Table 5-4) or 0.98% dry weight. These values are well below the 5% TOC (dry weight) criteria established by the organic safety program (Babad, Blacker, and Redus 1994). The waste has not been analyzed for cyanide. However, Tank BX-105 history and Borsheim and Simpson (1991) do not show that it should contain significant amounts of cyanide. The 1994 DSC results and the 1986 results do not indicate that the waste contains excessive amounts of fuel.

Large amounts of moisture reduce the potential for propagating exothermic reactions in the waste. For the 1994 auger sampling event, the average wt% water of all the TGA analyses was 12.6%. Although this is below the 17 wt% criteria identified in the safety screening DQO, there is no indication that a significant fuel source is present or that a significant potential for runaway reactions exists in the tank.

The potential for criticality can be assessed from either the total alpha results from the 1994 auger sampling event, or the plutonium results from the 1986 core sampling event. The highest total alpha result from the 1994 auger subsamples was 0.589 μCi/g. The highest plutonium concentration measured in the 1986 solids composites was 0.474 μCi/g. Assuming all the alpha emitters and plutonium in the waste consist of the ^{239}Pu isotope, using the specific activity for ^{239}Pu of 6.2 x 10^{2} Ci/g, and a waste bulk density of 1.5 g/mL, the maximum measured total alpha and plutonium concentrations convert to 1.43 x 10^{-2} g/L and 1.15 x 10^{-6} g/L, respectively. This is a conservative assumption. These values are well below the established criteria of 1 g/L specified in the safety screening DQO (by a factor of 70 or more).

Analysis of the tank head space for flammable gas was not conducted. Information on vapor space sampling and analysis will be included in a future revision to this report.

Another factor in assessing the safety of the tank waste is the heat generation and temperature of the waste. Based on the 1986 data, the heat generation rate for the tank is estimated to be 327 W (1120 Btu/hr) (see Table 5-4). This is far below the 40,000 Btu/hr criteria for distinguishing a high heat load tank from a low heat load tank (Hanlon 1995). The recorded tank temperature since 1991 has been between 25 °C and 35 °C.

Table 5-5 lists the DQO required analytes, their notification limits, and their analytical results.

5.5.2 Operational Evaluation

No analyses were performed for specific operational purposes. Total alpha data from the 1994 sampling event and plutonium analyses from 1986 indicate that the waste exceeds the Transuranic (TRU) (waste) criteria of 100 nCi/g. TOC levels are below normal operational concerns. The 1986 analyses for the riser 1 solids composite indicate that the waste may contain a very high concentration of phosphorus. A significant fraction of the phosphorus is water insoluble (see Table 5-6). Insoluble phosphates can be very hard, making pumping difficult. However, the fact that the 1986 core samples could be taken in push mode, and that the waste was penetrated to the bottom of the tank in push mode, indicates that the waste is probably of a consistency amenable to pumping. Other operational factors need to be considered as part of the assessment before the waste is transferred. However, waste should not need to be pumped since the tank is interim stabilized and intrusion prevention has been performed.

5.5.3 Process Development Evaluation

The results for the water soluble, acid soluble, and acid insoluble fractions from the 1986 testing will be important for evaluating the disposal waste form (glass) formulations and identifying potential components that may affect the treatment and disposal process. Because the waste sludges will probably be blended, washed and treated before disposal there are no specific criteria for the parameters measured. Extensive rheological analyses have yet to be conducted on the waste. Once these evaluations are performed, the results will assist the retrieval and pretreatment programs in determining equipment needs.

The metal analyses information reported on the water digested solids from the 1986 sampling event yields solubility information that may be useful in the retrieval, pretreatment, and vitrification of the tank waste. The solubility of several metals and radionuclides was examined by comparing the results from the water soluble fraction with the total concentration results from all fractions (water soluble, acid soluble, acid insoluble). Tables 5-6 and 5-7 list the percent solubility for several analytes from the riser 1 and 5 solids composites, respectively. Only those analytes that were detected in the leachate resulting from the water washing step were considered.

The data demonstrates that sodium and ^{137}Cs are highly water soluble, as expected. The phosporus in the tank was also found to be highly soluble (73.8 to 91.5 percent), although enough insoluble phosphate may exist to affect pumpability. The boron and ^{99}Tc species were found to be 100 percent water soluble. The potassium and ^{60}Co species were found to be relatively water soluble, ranging from 51.5 to 59.4 percent and 19.3 to 51.9 percent, respectively.

The remaining metals and radionuclides were found to be relatively insoluble. The chromium data indicate that chromium is present as the Cr(III) rather than the soluble Cr(VI) species.

Table 5-5. Safety Screening DQO Decision Variables and Criteria for Tank 241-BX-105.

Safety issue	Primary decision variable	Decision critera threshold	Analytical value
Ferrocyanide/ Organic	Total Fuel Content	481 J/g (115 cal/g)	No exotherms observed
Organic	Percent Moisture	17 wt %	12.6 wt %[a]
Criticality	Total Alpha	34.4 μCi/g (1 g/L)[b]	0.589 μCi/g[c]
Flammable Gas	Flammable Gas	< 25% LFL	ND

[a]Represents the average of the TGA results from both risers 2 and 6.

[b]Although the actual decision criterion listed in the DQO is 1g/L, total alpha is measured in μCi/g rather than g/L. To convert the notification limit for total alpha into a number more readily usable by the laboratory, it was assumed that all alpha decay originates from ^{239}Pu. Using the average bulk density value from Table 4-3 of 1.79 g/ml, the decision criterion may be converted to 34.4 μCi/g as shown:

$$\left(\frac{1\ g}{L}\right)\left(\frac{1\ L}{10^3\ mL}\right)\left(\frac{1}{\text{density}}\frac{mL}{g}\right)\left(\frac{0.0615\ Ci}{1\ g}\right)\left(\frac{10^6\ \mu Ci}{1\ Ci}\right) = \frac{61.5}{\text{density}}\frac{\mu Ci}{g} \qquad (1)$$

[c]Highest result from 1994 data.

Table 5-6. Percent Water Solubility for Riser 1 Solids Composite Analytes.

Analyte	Concentration, water soluble fraction	Concentration, all fractions	Percent soluble
Metals	(μg/g)	(μg/g)	(%)
Aluminum	1,660	16,800	9.88
Barium	2.15	2,440	8.8 E-01
Boron	6.09	6.09	100
Calcium	102	5,160	1.98
Chromium	158	14,700	1.07
Copper	2.63	32.5	8.09
Iron	2.58	9,800	2.63 E-02
Lead	39.5	831	4.75
Magnesium	2.79	2,620	0.106
Nickel	32	221	14.5
Phosphorus	23,600	32,000	73.8
Potassium	849	1,430	59.4
Silicon	979	33,300	2.94
Sodium	104,000	142,000	73.2
Strontium	0.762	70.5	1.08
Uranium	3.14	6,600	4.76 E-02
Radionuclides	(μCi/g)	(μCi/g)	(%)
$^{239/240}Pu$	1.89 E-02	0.474	3.99
^{90}Sr	2.83	227	1.25
^{99}Tc	3.63 E-02	3.63 E-02	100
^{241}Am	2.13 E-02	1.65	1.29
^{60}Co	4.99 E-02	0.259	19.3
^{137}Cs	49	62.3	78.7

Table 5-7. Percent Water Solubility for Riser 8 Solids Composite Analytes.

Analyte	Concentration, water soluble fraction	Concentration, all fractions	Percent soluble
Metals	(μg/g)	(μg/g)	(%)
Aluminum	1,200	50,900	2.36
Barium	1.93	3,950	4.89 E-02
Boron	4.52	4.52	100
Calcium	73.2	7,260	1.01
Chromium	145	2,530	5.73
Copper	2.97	22.0	13.5
Iron	1.84	2,080	8.85 E-02
Lead	4.55	296	1.54
Magnesium	5.17	3,900	0.133
Nickel	20.3	73.9	27.5
Phosphorus	7,180	7,850	91.5
Potassium	772	1,500	51.5
Silicon	453	49,700	0.911
Sodium	54,600	77,800	70.2
Strontium	0.666	41.9	1.59
Uranium	2.04	2,210	9.23 E-02
Radionuclides	(μCi/g)	(μCi/g)	(%)
$^{239/240}$Pu	3.25 E-03	8.47 E-02	3.84
^{90}Sr	1.93	45.9	4.20
^{99}Tc	3.06 E-02	3.06 E-02	100
^{60}Co	2.91 E-02	5.61 E-02	51.9
^{137}Cs	41.2	45.9	89.8

6.0 CONCLUSIONS AND RECOMMENDATIONS

The TGA analyses on the 1994 auger samples indicate that the water content of the waste are below the safety screening criteria of 17 wt%. However, the DSC analyses on the auger samples did not show any measureable exothermic activity. Thermocouple temperature measurements in the tank and estimates of heat generation do no indicate any excessive heat sources. TOC results for the sludge analyzed in 1986 do not indicate that a large fuel source is present. Even though the water content is low, the absence of fuel indicates that the potential for runaway reactions in the tank is unlikely.

The total alpha results from 1994 and the plutonium results from 1986 indicate that the plutonium concentration is well below the safety screening criticality criteria. However, the results indicate that the plutonium and americium concentrations are above the TRU classification limit of 100 nCi/g. The large range in results between risers indicates a potential for large variability in the plutonium concentration in the waste.

The 1986 analyses indicate that the sludge contains relatively large amounts of sodium, aluminum, chromium, iron, phosphorus, silicon and NO_3. There is limited analytical data on the concentration of other anions besides nitrate. High concentrations of sodium, NO_3, and phosphates are expected in the tank based on its fill history, which includes metal waste from the bismuth phosphate process employed at B Plant in the 1940's and 1950's and uranium recovery waste in 1956. The high aluminum and NO_3 concentration is also expected based on the receipt of PUREX cladding waste in the 1960's. The major radioisotopes in the waste are $^{239/240}Pu$, ^{241}Am, ^{90}Sr, and ^{137}Cs. Like the plutonium, several of these isotopes show a large range in concentration between samples taken from the two risers on opposite sides of the tank. The variability in the analyte concentrations between risers indicates that the tank waste may exhibit lateral, as well as vertical, heterogeneity.

Sample recovery from risers 1 and 2 in 1986 and 1994, respectively, were poor. Sampling another riser to improve waste recovery, to better evaluate the variability of the waste composition in the tank, and to verify the wt% water results of the 1994 auger samples should be considered. If possible, future samples should be taken from the center riser on the tank. Samples to date were taken near the perimeter of the tank, where the waste composition may be substantially different than near the center. For example, it is likely that the waste near the edges is drier based on the in-tank photographs. Sampling near the center will also allow samples to be taken of the heel material at the bottom of the dish. Extensive anion analysis and TOC analysis should be performed on any archived or future samples to provide a more accurate estimate of the waste composition. Measures should be taken in any future core sampling to minimize the potential for contamination due to NPH hydrostatic fluid.

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APPENDIX A

ANALYTICAL PROCEDURE NUMBERS
SINGLE SHELL TANK 241-BX-105

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Table A-1. Summary of Analytical Methods, 1986 Core Sampling Event. (2 sheets)

Analyte/property	Procedure number
Density (liquids)	LA-510-112
Aluminum	LA-505-143
Barium	LA-505-143
Bismuth	LA-505-143
Boron	LA-505-143
Cadmium	LA-505-143
Calcium	LA-505-143
Cobalt	LA-505-143
Copper	LA-505-143
Iron	LA-505-143
Lead	LA-505-143
Magnesium	LA-505-143
Manganese	LA-505-143
Nickel	LA-505-143
Phosphorus	LA-505-143
Potassium	LA-505-143
Silicon	LA-505-143
Silver	LA-505-143
Sodium	LA-505-143
Strontium	LA-505-143
Zinc	LA-505-143
Zirconium	LA-505-143
Uranium	LA-925-106
Nitrate	LA-533-103

Table A-1. Summary of Analytical Methods, 1986 Core Sampling Event. (2 sheets)

Analyte/property	Procedure number
TOC	LA-344-101
pH	LA-212-102
Water loss	LA-560-111
$^{239/240}Pu$	LA-503-155
^{14}C	LA-348-102
^{90}Sr	LA-220-101
^{99}Tc	LA-430-101
^{241}Am	LA-503-155
^{60}Co	LA-540-121
^{137}Cs	LA-540-121
^{129}I	No Procedure
Bulk Density (solids)	LA-560-101

APPENDIX B

HEAT LOAD ESTIMATE
SINGLE-SHELL TANK 241-BX-105

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Table B-1. Heat Load Estimate, Tank 241-BX-105.

Radionuclide	Total tank inventory (Ci)	Decay heat generation rate (W/Ci)	Heat load (W)
^{241}Am	226	3.28 E-02	7.41
^{60}Co	46.1	1.54 E-02	0.709
^{137}Cs	2.25 E+04	4.72 E-03[a]	106
$^{239/240}Pu$	64.1	3.06 E-02	1.96
^{90}Sr[b]	3.15 E+04	6.69 E-03	211
^{99}Tc	14.3	5.01 E-04	7.15 E-02
Total Heat Load:			327

[a]Decay heat includes ^{137}Ba, which is in equilibrium with ^{137}Cs.
[b]Decay heat includes $^{Y}90$, which is in equilibrium with ^{90}Sr.

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